Question # 1

Given the following:

- Unit 4 is at 100% power.

Subsequently:

- The following alarms are received (NOT all inclusive):
    - ANN A9/5, PZR PRESSURE CONTROLLER HIGH OUTPUT.
    - ANN A4/1, PORV/ SAFETY VALVE OPEN.
    - ANN A7/2, PZR PORV HI TEMP.

- The RCO MANUALLY closes:
    - PCV-4-455A, PZR Spray Loop C.
    - PCV-4-455B, PZR Spray Loop B.
    - PCV-4-455C, PZR PORV.

Which one of the following completes the statements below?

To address this transient the crew will enter __(1)__.

To confirm that the PORV(s) is(are) fully closed the operator will
Monitor a PORV Tailpipe Temperature Indicator that is __(2)__.

A. (1) 4-ONOP-41.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE
(2) common for both PORVs

B. (1) 4-ONOP-41.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE
(2) separate for each PORV

C. (1) 4-ONOP-041.5, PRESSURIZER PRESSURE CONTROL MALFUNCTION
(2) common for both PORVs

D. (1) 4-ONOP-041.5, PRESSURIZER PRESSURE CONTROL MALFUNCTION
(2) separate for each PORV

Answer Analysis

Discussion: PC-444J PZR PRESSURE CONTROL failing high will cause the two spray valves to open and above 92% cause PZR PORV 455C to open

A. Incorrect.
Part 1 incorrect, plausible since since some RCS inventory is lost to the PRT During this transient.
Part 2 correct.

B. Incorrect.
Part 1 incorrect, plausible since since some RCS inventory is lost to the PRT During this transient.
Part 2 incorrect, plausible since since each of the RCS Safety Valves have their individual tailpipe temperature indication.

C. Correct. For PZR pressure control malfunctions 4-ONOP-041.5 is the correct procedure. Both PORVs have a common tailpipe temperature indicator.

D. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since since each of the RCS Safety Valves have their individual tailpipe temperature indication.

Tier: 1 **Group:** 1

K/A: 008K.2.02; /; Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Sensors and detectors

Importance Rating: **2.7**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must evaluate the annunciator provided and the operator procedure required to mitigate the event. The applicant must also demonstrate fundamental knowledge to realize one of the PORVs is still open as the temperature is still above saturation pressure for the PORV. Thus the temperature detector is indicating correctly that the PORV is still open

SRO Justification: N/A

Technical References: **4-ONOP-041.5**- Pressurizer Pressure Control Malfunctions, steps 1 and 2. UFSAR section 14.1.14 Accidental Depressurization of the Reactor Coolant System. 5614-M-3041 SH 2.

Proposed references to be provided: None

Learning Objective: Lesson Plan 6902204 Pressurizer Pressure Control Malfunction

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

Comments: RO question.

- Tier match: Question is related to a procedure 4**-ONOP-041.5**- Pressurizer Pressure Control Malfunctions.
- RO only: Answer options require knowledge of only system behavior, sequence, and mitigation strategy of procedure.
- K/A Match: UFSAR does discuss PZR Spray valves as one cause of Accidental Depressurization of the Reactor Coolant System event

1.0 PURPOSE

1.1 This procedure provides operator response guidelines for pressure control in the event either normal pressurizer pressure control malfunctions or a PZR PORV fails when aligned for normal pressure control.

2.0 SYMPTOMS

2.1 Annunciators

2.1.1 Annunciator A 9/2, PZR CONTROL HI/LO PRESS

2.1.2 Annunciator A 9/5, PRZ PRESSURE CONTROLLER HI OUTPUT

2.1.3 Annunciator A 4/1, PORV/SAFETY OPEN

2.1.4 Annunciator A 7/1, PRT HI/LO LEVEL HI PRESS/TEMP

2.1.5 Annunciator A 7/2, PZR PORV HI TEMP

2.1.6 Annunciator A 7/3, PZR SAFETY VALVE A/B/C HI TEMP

2.2 Indications

2.2.1 PZR relief line temperature, TI-4-463, INCREASING

2.2.2 PZR relief tank level, LI-4-470, INCREASING

2.2.3 PZR relief tank temperature, TI-4-471, INCREASING

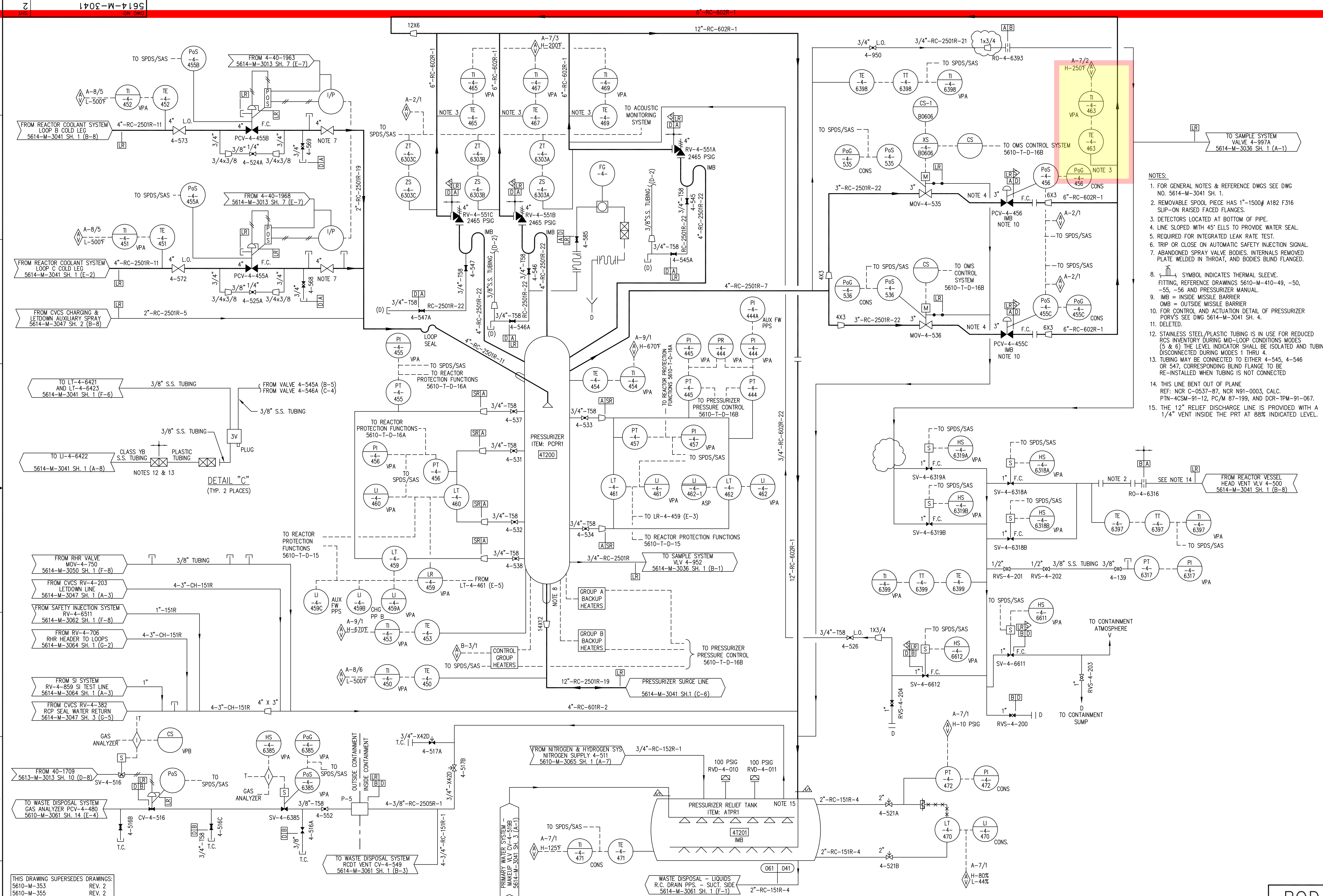
2.2.4 PZR relief tank pressure, PI-4-472, INCREASING

2.2.5 PZR PORV/Safety Acoustic Monitor, LEDs LIT

2.2.6 PZR PORV indicating lights indicate OPEN OR INTERMEDIATE

2.2.7 PZR PORV Block Valve indicating lights indicate INTERMEDIATE

2.2.8 Pressurizer spray valves, heaters and/or PORVs operation not compatible with plant conditions.



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5614-M-3041 SH. 1.
2. REMOVABLE SPOOL PIECE HAS 1"-1500# A182 F316 SLIP-ON RAISED FACED FLANGES.
3. DETECTORS LOCATED AT BOTTOM OF PIPE.
4. LINE SLOPED WITH 45° ELLS TO PROVIDE WATER SEAL.
5. REQUIRED FOR INTEGRATED LEAK RATE TEST.
6. TRIP OR CLOSE ON AUTOMATIC SAFETY INJECTION SIGNAL.
7. ABANDONED SPRAY VALVE BODIES. INTERNALS REMOVED PLATE WELDED IN THROAT, AND BODIES BLIND FLANGED.
8. SYMBOL INDICATES THERMAL SLEEVE. FITTING, REFERENCE DRAWINGS 5610-M-410-49, -50, -55, -56 AND PRESSURIZER MANUAL.
9. IMB = INSIDE MISSILE BARRIER
   OMB = OUTSIDE MISSILE BARRIER
10. FOR CONTROL AND ACTUATION DETAIL OF PRESSURIZER PORV'S SEE DWG 5614-M-3041 SH. 4.
11. DELETED.
12. STAINLESS STEEL/PLASTIC TUBING IS IN USE FOR REDUCED RCS INVENTORY DURING MID-LOOP CONDITIONS MODES (5 & 6) THE LEVEL INDICATOR SHALL BE ISOLATED AND TUBING DISCONNECTED DURING MODES 1 THRU 4.
13. TUBING MAY BE CONNECTED TO EITHER 4-545, 4-546 OR 547, CORRESPONDING BLIND FLANGE TO BE RE-INSTALLED WHEN TUBING IS NOT CONNECTED
14. THIS LINE BENT OUT OF PLANE REF: NCR C-0537-87, NCR N91-0003, CALC. PTN-4CSM-91-12, PC/M 87-199, AND DCR-TPM-91-067.
15. THE 12" RELIEF DISCHARGE LINE IS PROVIDED WITH A 1/4" VENT INSIDE THE PRT AT 88% INDICATED LEVEL.

THIS DRAWING SUPERSEDES DRAWINGS:		
5610-M-353		REV. 2
5610-M-355		REV. 2
5610-M-366		REV. 4
5610-M-367		REV. 2
5610-M-410-91	SH. 1	REV. 13
5610-M-410-91	SH. 2	REV. 30
5610-M-410-202	SH. 1	REV. 4
5610-M-410-202	SH. 2	REV. 6
5610-M-1288		REV. 4
5610-M-1310		REV. 2

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4501 SH. 1 REV. 90

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
32	4-27-05	ISSUED AS-BUILT PER PC/M 04-065 & INCORP. CRN-M-11240.	RH	RV	BSG	JTL	38	2/6/13	ISSUED AS-BUILT PER EC 242470 (ITOP 13-02-027).	RH	AFG	–	DL
31	4-21-05	ISSUED AS-BUILT PER PC/M 04-056.	RH	JPG	AAP	CMZ	37	9/28/09	ISSUED AS-BUILT PER CRN-I-5092 (PC/M 09-070).	RV	RH	–	LG
42	4-7-15	ISSUED AS-BUILT PER EC 283531.	RH	BB	–	SB	36	9-22-09	ISSUED AS-BUILT PER CRN M-12837 (PC/M 09-070).	RH	RV	–	LG
41	4-24-13	ISSUED AS-BUILT PER EC 246978 (ITOP 13-04-164).	RH	RV	–	JM	35	9/25/06	ISSUED AS-BUILT PER CRN-M-11759 (PC/M 03-106).	RH	RV	BSG	JTL
40	3-4-13	ISSUED AS-BUILT PER EC 246924 (ITOP 13-03-010).	RV	RH	–	JD	34	7-11-05	ISSUED AS-BUILT PER CRN-M-11395 (PC/M 03-106).	RH	RRW	AAP	PRB
39	2-10-13	ISSUED AS-BUILT PER EC 246924 (ITOP'S 13-02-043 & 047). (PARTIAL)	RV	BB	PSB	CW	33	6-28-05	ISSUED AS-BUILT PER CRN M-11337 (PC/M 04-065).	RH	RV	RSV	ASD

SAFETY RELATED

POD

TURKEY POINT NUCLEAR UNIT 4

P & I D

REACTOR COOLANT SYSTEM

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5614-M-3041

SHEET 2

SYS 041

REV 42

5614-M-3041_SH 2.DWG

Question # 2

Given the following:

- Unit 3 is at 100% power.

Subsequently:

- ANN G1/2, CHARGING PUMP HI SPEED, alarms.
- 3-ONOP-041.3, Excessive Reactor Coolant System Leakage, is entered.
- PZR level is lowering at 2%/min.

Which one of the following completes the statements below?

Assuming no operator manipulation, PZR heaters will DE-ENERGIZE within __(1)__ from event initiation at the earliest.

The foldout page for 3-ONOP-041.3, __(2)__ ~~applicable~~ required to be monitored during the performance of step 1, “Maintain RCS inventory”.

A. (1) 21.5 minutes
   (2) is

B. (1) 21.5 minutes
   (2) is NOT

C. (1) 25.5 minutes
   (2) is

D. (1) 25.5 minutes
   (2) is NOT

Answer Analysis

The normal program PZR Level at 100% power is 57% (Reference drawing 5610-T-D-15). The candidate needs to understand this is the value for normal full load operations for a starting point to determine the time until Pressurizer Heaters automatically deenergize at 14%.

A. (1) Correct. (57%-14%)/2%/min= 21.5 minutes
   (2) Correct. See Page 7 of 3-ONOP-041.02

B. (1) Correct. See A(1)
   (2) Incorrect. Plausible if the applicant does not understand the use of foldout requirements in off normal procedures.

C. (1) Incorrect. Plausible if candidate does not understand that the PZR Heaters will receive a signal to deenergize at the same time as the PZR Low Level alarm at 14%. At 6% there is the PZR Low-Low level alarm that could be confused with the PZR deenergize setpoint. (57%-6%)/ 2%/min = 25.5 min.
   (2) Correct. See A(2)

D. (1) Incorrect. See C(1)
   (2) Incorrect. See B(2)

Question Number: 2

Tier: **1** **Group:** 1

K/A: 009EA2.05
Ability to determine or interpret the following as they apply to a small break LOCA:
The time available for action before PZR is empty, given the rate of decrease of PZR level

Importance Rating: **3.4**

10 CFR Part 55: 45.13

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, identify automatic system response.

SRO Justification: N/A

Technical References: 3-ONOP-041.3, 3-OP-047, 5610-T-D-19

Proposed references to be provided: None

Learning Objective: PTN 6902113 OBJ. 6

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Foldout Page SHALL be monitored throughout this procedure.

1 **Maintain RCS Inventory**

a. Maintain RCS Inventory as directed by the Unit Supervisor

* Maintain program level

OR

* Maintain ordered band for operational mode

OR

* Maintain unit water solid (if unit water solid prior to event)

b. Start additional charging pumps as necessary to maintain RCS Inventory

c. **IF** charging flow is maximum, **THEN** isolate letdown flow

NOTE

RCS Inventory is either Prz Level, Drain Down Level, or Cavity Level.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	**Check RCS Inventory Decreasing**	Go to Step 10.
3	**Check The Following** a. Charging flow - MAXIMUM b. Letdown flow - ISOLATED	Return to Step 1.
4	**Check Unit In Mode 1 Through 3 Greater Than 1000 psig With Safety Injection System Aligned For Injection**	Go to Step 6.
5	**Manually Trip The Reactor And Go To 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION**	
6	**Check Unit Operating Mode 3 Less Than 1000 psig With Safety Injection Blocked Or Mode 4**	Go to Step 8.

INITIALS
CK'D VERIF

7.20.2 (Cont'd)

NOTES

- *An approximate 450 to 500 gallon blend should be sufficient to achieve VCT level indication. Once on scale, an additional 450 to 550 gallons should be sufficient to achieve adequate VCT level.*
- *VCT level is 14.15 gallons per% level indication.*
- *VCT make up flowrate should be low to ensure proper system response (e.g. 15 to 30 gpm Primary Water and 10 to 20 gpm Boric Acid) and PI-3-117 shall be monitored while fill is in progress.*
- *Adjustments to Boric Acid flow rate may be necessary to ensure all of the required Boric Acid is injected prior to the completion of the manual make up.*

3. **IF** it is desired to fill the VCT via the CVCS Boron Concentration System, **THEN** perform the following:

____ a. Determine the approximate boric acid and primary water flows and volumes needed to obtain the desired blend concentration from the boron change tables in Section III of the Plant Curve Book. The boric acid flow rate should be adjusted in order to ensure all of the boric acid is injected prior to the completion of the manual make up.

____ ____ b. Open VCT Inlet Stop Valve, 3-247B.

____ ____ c. Open VCT Gas Sample Line Drain Valve, 3-976.

____ ____ d. Place VCT Divert to Hold Up Tank, LCV-3-115A to VCT position.

____ e. Adjust Boric Acid Flow Controller, FC-3-113A Auto Setpoint to the value determined in Substep 7.20.2.3.a.

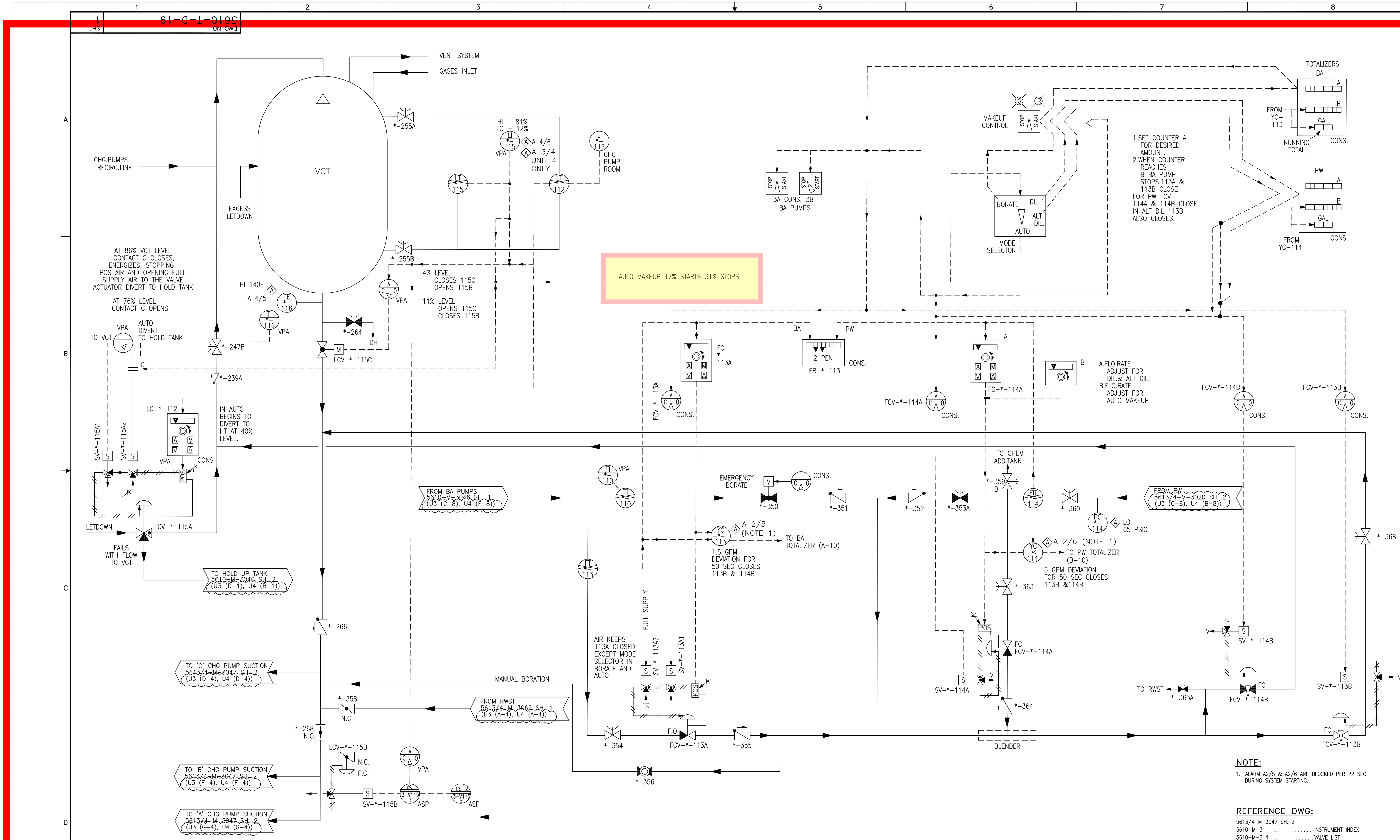
____ f. Place Primary Water Flow Controller, FC-3-114A in Manual and adjust the output on the demand meter to 0%.

____ g. Place the Reactor Makeup Selector Switch to Borate.

____ h. Ensure Control Switch for Boric Acid to Blender, FCV-3-113A is in Auto.

____ i. Place Control Switch for Blender to Charging Pp Suction, FCV-3-113B to Close.

____ j. Place Control Switch for Blender to VCT, FCV-3-114B to Open.



REV	DATE	REVISION	BY	CH	APP	APP
17	2-20-12	ISSUED AS-BUILT PER EC 242547 (PC/M 10-010).	RV	RH	–	LG
16	4-7-11	ISSUED AS-BUILT PER EC 242547 (PC/M 10-010).	RV	SB	–	LG
15	4-1-10	ISSUED AS-BUILT PER CRN-E-18146 (PC/M 10-007).	RV	RH	–	LG

REV	DATE	REVISION	BY	CH	APP	APP
14	10/5/06	ISSUED AS-BUILT PER CRN-I-4476 (PC/M 03-106).	RH	JPG	–	PDS
13	2-18-04	ISSUED AS-BUILT PER CRN-I-4167(PC/M 03-106).	RV	RH	TBH	TS
12	4-5-95	ISSUED AS-BUILT FOR PC/M'S 95-002 & 95-003.	RV	RH	BAW	JCW
11	3-18-94	ISSUED AS-BUILT FOR PC/M 93-216.	RV	RAS	–	JCW
10	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	GM	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

CONTROL SYSTEM DIAGRAM

BORON CONCENTRATION CONTROL SYSTEM

POD

DRAWING NUMBER 5610-T-D-19 SYS – REV 17 SHEET 1

5610-T-D-19_SH 1.DWG

Question # 3

Given the following sequence of events:

Time 1400:

- Unit 3 is at 100% power.
- Charging pump 3A is running with abnormal noise, flow oscillations and high vibrations.

Time 1410:

- ANN A 5/1, CHARGING PUMP A TRIP, alarms.
- 3-ONOP-047.1, Loss of Charging Flow in Modes 1 Through 4, is entered.

Which one of the following completes the statements below?

IAW 3-ONOP-047.1, a plant shutdown using 3-GOP-100, Fast Load Reduction __(1)__ desired.

If 3-ONOP-047.1, 3-EOP-E-0 Transition Criteria is met the crew will __(2)__ and transition to 3-EOP-E-0, Reactor Trip or Safety Injection.

A. (1) is
(2) ONLY trip the reactor

B. (1) is
(2) trip the reactor, initiate SI and Phase A

C. (1) is NOT
(2) ONLY trip the reactor

D. (1) is NOT
(2) trip the reactor, initiate SI and Phase A

Discussion:

A. Incorrect.
Part 1 incorrect, plausible since this is the course of actions for other ONOPs, if the contingency actions for the ONOP are not successful a Fast Load Reduction is performed, i.e. ONOP-014, Loss of Condenser Vacuum.
Part 2 incorrect, plausible since in other ONOPs that require a reactor trip, will NOT actuate SI and PH A, i.e . 3-ONOP-041.1, RCP Malfunction.

B. Incorrect.
Part 1 incorrect, plausible since this is the course of actions for other ONOPs, if the contingency actions for the ONOP are not successful a Fast Load Reduction is performed, i.e. ONOP-014, Loss of Condenser Vacuum.
Part 2 correct.

C. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since in other ONOPs that require a reactor trip, will NOT actuate SI and PH A, i.e . 3-ONOP-041.1, RCP Malfunction.

D. Correct. IAW 3-ONOP-047.1.

Tier: 1 **Group:** 1

K/A: 0022AA2.04; /; **Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup:** Charging pump problem

Importance Rating: **2.9**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must evaluate the charging pump symptoms presented and then have knowledge of the 0NOP overall mitigation sequence and strategy.

SRO Justification: N/A

Technical References: ARP A 5/1 and A 9/3 (Charging pump trip and PZR level deviation)
3-ONOP-047.1 - Loss of Charging Flow

Proposed references to be provided: None

Learning Objective: Lesson Plan 6910113 CVCS system
6910109 Pressurizer

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *A plant shutdown using either 3-GOP-100, FAST LOAD REDUCTION or 3-GOP-103, POWER OPERATION TO HOT STANDBY, is NOT desired with a loss of charging because of the high potential of being in an at-power RIL Lo-Lo rod insertion unanalyzed condition.*

- *Foldout Page should be reviewed before commencing this procedure.*

1 **Check Any Charging Pumps Running**

a. Perform the following to start a charging pump:

1) Verify VCT level, LT-3-115, greater than 4% AND LCV-3-115C Open.

2) **IF** unable to open LCV-3-115C, **THEN** Open LCV-3-115B.

3) **IF** unable to open LCV-3-115B, **THEN** locally Open 3-358, RWST Emer Makeup to Chrg Pumps LCV-3-115B Bypass.

4) **IF** there is a problem with the VCT Level Control System, **THEN** refer to 3-ONOP-046.4, MALFUNCTION OF BORON CONCENTRATION CONTROL SYSTEM.

b. Verify Open Charging Flow to Regen Hx, HCV-3-121.

c. Verify Open Loop A Charging Isolation, CV-3-310A.

d. Start functional charging pumps as necessary to restore pressurizer level.

e. Adjust charging pump speed controllers to restore pressurizer level to program.

f. Go to Step 2.

FOLDOUT PAGE FOR 3-ONOP-047.1

1. **ADVERSE CONTAINMENT CONDITIONS**

IF either of the conditions listed below occur, **THEN** use [Adverse Containment Setpoints]:

* Containment atmosphere temperature greater than or equal to 180°F

OR

* Containment radiation levels greater than or equal to $1.3x10^5$ R/hr

WHEN Containment atmosphere temperature returns to less than 180°F,
THEN Normal Setpoints can again be used.

WHEN Containment radiation levels return to less than $1.3x10^5$ R/hr,
THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.

2. **3-EOP-E-0 TRANSITION CRITERIA**

IF PZR level is 10% below program **OR** can **NOT** be maintained above 7%, **THEN** perform the following:

- Trip the Reactor and Turbine.
- Initiate Safety Injection **AND** Phase A Containment Isolation.
- Go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

3. **3-ONOP-041.7 TRANSITION CRITERIA**

IF PRZ level can **NOT** be maintained above 7% with the plant in Mode 3 (less than 1000#), or Mode 4,
THEN go to 3-ONOP-041.7 SHUTDOWN LOCA[Mode 3 (Less than 1000 psig) or Mode 4].

4. **RESTORATION OF CHARGING**

IF charging capability is restored any time during the performance of this procedure, **THEN** perform the following, if desired:

a. Reestablish letdown using ATTACHMENT 1.
b. Reestablish charging using ATTACHMENT 2.
c. Establish pressurizer level on program.
d. Go to plant procedure appropriate for plant conditions.

5. **TECH SPEC MONITORING**

Monitor Tech Spec 3.1.2.2 and 3.1.2.3 during the performance of this procedure.

Question # 4

Given the following:

- Unit 4 plant cooldown to MODE 5 is in progress.
- Both RHR pumps are running.
- OMS is in Low Pressure Ops.
- PZR is solid.

Subsequently:

- ANN A3/3, OMS CONTROL ACTIVATED, alarms.
- ANN H7/1, RHR PP A HI PRESS, and ANN H7/2, RHR PP B HI PRESS alarms.
- The crew enters, 4-ONOP-050, Loss of RHR.

Which one of the following completes the statements below?

IAW 4-ONOP-050, the crew will __(1)__ when RCS pressure lowers, AT LEAST, below a setpoint of __(2)__.

A. (1) Re-start ONE Charging Pump
(2) 525 psig

B. (1) Re-start ONE Charging Pump
(2) 440 psig

C. (1) Re-open MOV-4-750 and MOV-4-751, RHR Suction Valves
(2) 525 psig

D. (1) Re-open MOV-4-750 and MOV-4-751, RHR Suction Valves
(2) 440 psig

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. Incorrect.
   Part 1 incorrect, plausible since a running Charging Pump will raise RCS pressure and it is plausible that the OMS system would trip the running Charging Pumps. Also ONOP-050 includes steps for securing Charging Pumps if pressure is > 525 psig.
   Part 2 correct.

B. Incorrect.
   Part 1 incorrect, plausible since a running Charging Pump will raise RCS pressure and it is plausible that the OMS system would trip the running Charging Pumps. Also ONOP-050 includes steps for securing Charging Pumps if pressure is > 525 psig.
   Part 2 incorrect, plausible since this is the OMS actuation setpoint.

C. Correct. 4-ARP-097.CR.H, Rev 11 describes the actuation setpoint (>550#) for the alarms and that no automatic actions are associated with the alarm. The note describes the automatic action associated >525 #. Therefore during a pressure excursion, the valves will have started to close before reaching the 550# alarm setpoint. 4-ONOP-050 will re-open the RHR Suction Valves when the RCS pressure lowers below 525 psig.

D. Incorrect.
   Part 1 correct.
   Part 2 incorrect, plausible since this is the OMS actuation setpoint.

Question Number: 4

Tier: 1 **Group:** 1

K/A: 000025 (APE 25) Loss of Residual Heat Removal System
025G2.4.46; Ability to verify that the alarms are consistent with the plant conditions.

Importance Rating: **4.2**

10 CFR Part 55: 41.10

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-ONOP-050

Proposed references to be provided: None

Learning Objective: 6902210 Obj. 4

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Interrupt feature for MOV-3-750 and MOV-3-751 is functional only with OMS in LO PRESS OPS.

2 **Check Loop 3C RHR Pump Suction Stop Valves – OPEN**

- MOV-3-750
- MOV-3-751

Perform the following:

a. Stop RHR pumps.

b. **IF** a momentary pressure spike has caused either or both valves to start closing, **THEN** perform the following at the Pushbutton Interrupt switches:

1) Determine affected valve(s).
   - Yellow light – ON
2) Verify over pressure signal **NOT** present:
   - Blue light – ON
3) Push Interrupt Pushbutton for affected valve(s).
4) Verify yellow light - DE-ENERGIZES.
5) **WHEN** blue light DE-ENERGIZES, **THEN** verify affected valve(s) - OPEN.
6) **IF** both valves are open, **THEN** go to Step 3.

c. **IF** RCS pressure GREATER THAN 525 psig, **THEN** perform the following:

1) Stop the charging pump(s).
2) Reduce RCS pressure to 425 psig.

d. **IF** MOV-3-750 and MOV-3-751 were **NOT** closed to isolate system leakage, **THEN** reopen MOV-3-750 and MOV–3-751. **IF** either valve can **NOT** be opened, **THEN** direct an operator to locally reopen MOV-3-750 and MOV-3-751.

e. **IF** BOTH valves can **NOT** be reopened, **THEN** monitor RCS Heatup Rate using Step 4 **AND** go to Step 11.

Question # 5

Given the following:

- Unit 3 is in MODE 1.
- 3D 4kv bus is being powered by 3A 4kv Bus.
- 3A CCW Pump is running.

Subsequently:

- 3A CCW Pump trips on overload.

Which one of the following completes the statements below?

LCO 3.7.2, Component Cooling Water System, __(1)__ currently satisfied.

NEXT the __(1)__ CCW Pump is expected to AUTO start.

A. (1) is
(2) 3B

B. (1) is
(2) 3C

C. (1) is NOT
(2) 3B

D. (1) is NOT
(2) 3C

Answer Analysis

A. Incorrect.
Part 1 incorrect, plausible since TS 3.7.2 is not required to be entered on inoperability of other CCW Components, i.e. CCW Heat Exchangers.
Part 2 correct.

B. Incorrect.
Part 1 incorrect, plausible since TS 3.7.2 is not required to be entered on inoperability of other CCW Components, i.e. CCW Heat Exchangers.
Part 2 incorrect, plausible since 3C CCW Pump is capable or autostarting on low pressure.

C. Correct. TS 3.7.2 requires all CCW Pumps Operable, hence this LCO is not met. 3B CCW will start NEXT since its time delay for starting on low pressure is shorter than 3C.

D. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since 3C CCW Pump is capable of autostarting on low pressure.

Tier: 1 **Group:** 1

K/A: 026G2.2.22; Knowledge of limiting conditions for operations and safety limits associated with Loss of Component Cooling Water / 8

Importance Rating: **4.0**

10 CFR Part 55: 41.5

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must evaluate the loss of a single CCW pump and have knowledge of the impact of a shared power supply and interlocks. Then the applicant evaluates the associated Tech Specs LCO (Above the line procedural aspect) .

SRO Justification: N/A

Technical References: TS 3.7.2
5610-T-L1 SH 24D

Proposed references to be provided: None

Learning Objective: PTN 6902140 OBJ. 6

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps, and

b. Two CCW heat exchangers.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

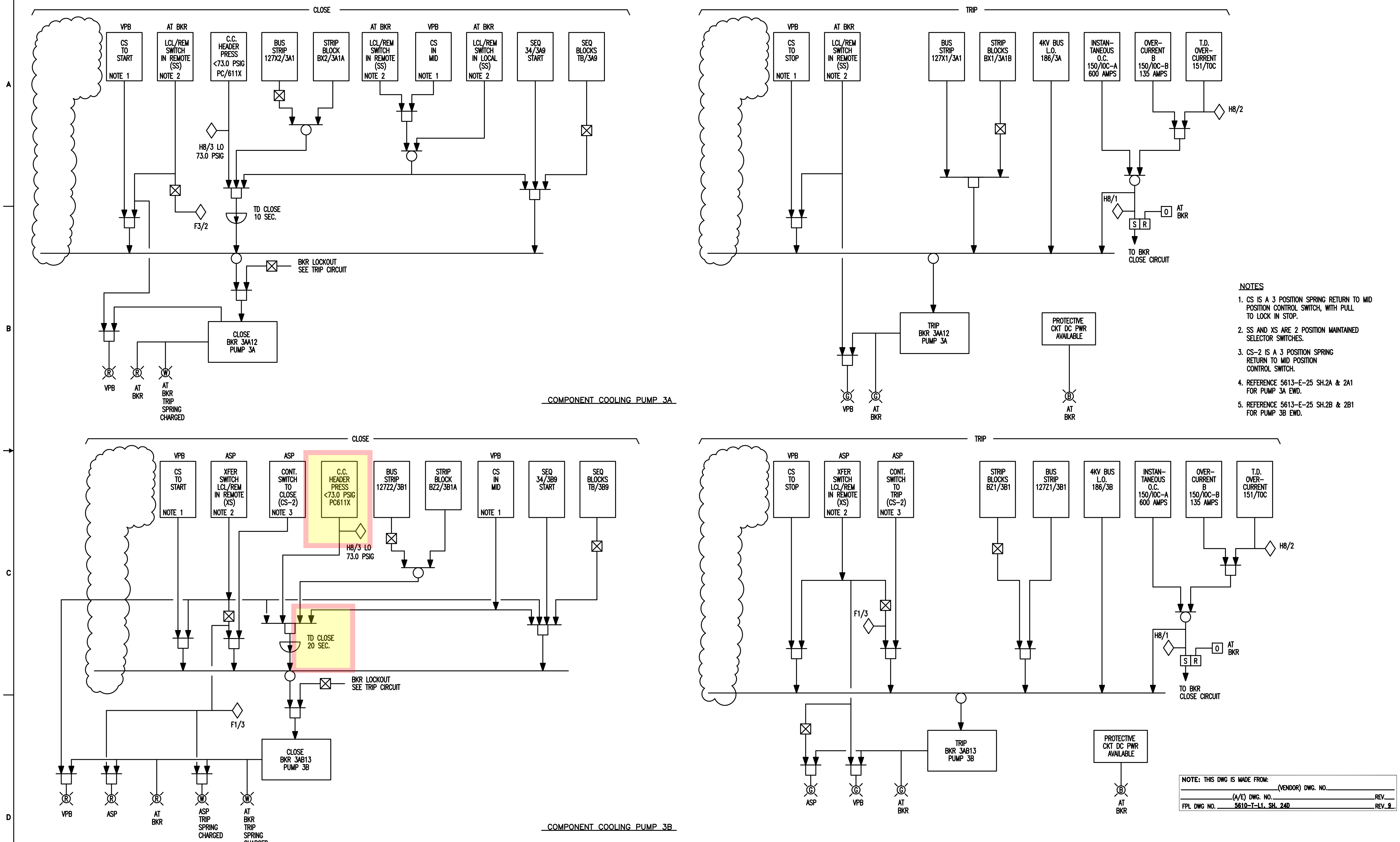
b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With less than two CCW heat exchangers OPERABLE, restore two heat exchangers to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.



REV	DATE	REVISION	BY	CH	APP	APP
14	12-18-13	ISSUED AS-BUILT PER EC-DCR 280474.	JJW	BB	–	MS
13	2-3-99	ISSUED AS-BUILT PER CRN-I-3786(PC/M 98-045).	RV	RH	BP	TS

REV	DATE	REVISION	BY	CH	APP	APP
12	10-20-98	ISSUED AS-BUILT PER PC/M 96-092.	RV	RH	BP	BAP
11	9-28-95	ISSUED AS-BUILT PER PC/M 95-130.	RV	RH	AD	JCW
10	9-22-95	ISSUED AS-BUILT PER PC/M 95-032.	RV	RH	TS	JCW
9	5-26-94	ISSUED AS-BUILT FOR PC/M 94-020 AND INCORP. CRN-I-1166.	RV	RH	JJN	JCW
8	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	GM	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNIT 3

LOGIC DIAGRAM

COMPONENT COOLING PUMPS 3A AND 3B

POD

DRAWING NUMBER 5610-T-L1

SYS 030

REV 14

SHEET 24D

5610-T-L1_SH 24D.DWG

Question # 6

Given the following:

- Unit 4 is at 100% power.

Subsequently:

- 4B Condensate Pump trips.
- PC-4-444J, Pressurizer Pressure Controller, does NOT respond in AUTOMATIC.
- ANN A9/3, PZR CONTROL HI/LO LEVEL, alarms.

Which one of the following completes the statements below?

Back-up Group Heaters __(1)__ INITIALLY energize.

IAW 4-ONOP-089, Turbine Runback, the RO will
INITIALLY __(2)__ the demand on PC-4-444J.

A. (1) will
(2) raise

B. (1) will NOT
(2) raise

C. (1) will
(2) lower

D. (1) will NOT
(2) lower

Answer Analysis

A. Incorrect.
   Part 1 incorrect, plausible since there is a signal that will send a trip signal to the PZR heaters on PZR level deviation.
   Part 2 correct.

B. Correct. The loss of the Feed pump results in a Turbine runback, which cause an insurge into the Pressurizer and an increase in pressure. The RCO will raise demand output on PC-4-444J to lower Pressurizer pressure (reduce Control Heater Output and open Spray). With increased RCS pressure Backup Heaters would NOT be on.

C. Incorrect.
   Part 1 incorrect, plausible since there is a signal that will send a trip signal to the PZR heaters on PZR level deviation.
   Part 2 incorrect, plausible since depending on the way components operate raising or lowering demand could cause different effects.

D. Incorrect.
   Part 1 correct.
   Part 2 incorrect, plausible since depending on the way components operate raising or lowering demand could cause different effects.

Question Number: 6

Tier: 1 **Group:** 1

K/A: 000027 (APE 27) Pressurizer Pressure Control System Malfunction.
027AA2.10; Ability to determine and interpret the following as they apply to
the Pressurizer Pressure Control Malfunctions:
PZR Heater Energized/De-energized Condition.

Importance Rating: **2.8**

10 CFR Part 55: XXXX

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 5610-T-L1 SH 23. 5610-T-D SH 16B. 4-ONOP-089

Proposed references to be provided: None

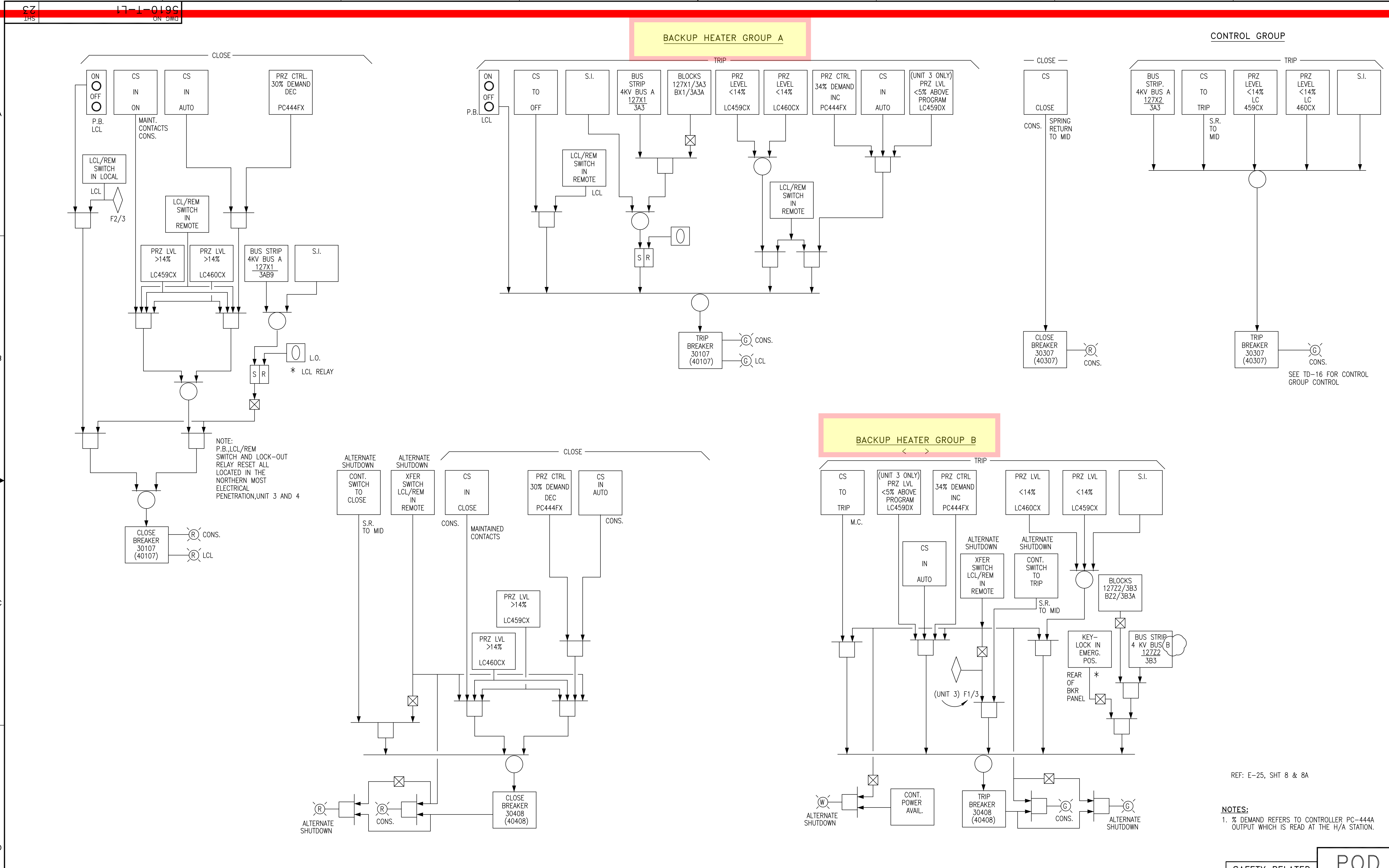
Learning Objective:

Cognitive Level: Higher X _
Lower _

Question Source: New _
Modified Bank X
Bank _

Question History: Modified from RO Question #45 from the 2015 NRC Exam.

Comments: Screenshot of development references as applicable



REV	DATE	REVISION	BY	CH	APP	APP
23	09-01-16	ISSUED AS-BUILT PER EC 286859.	AFG	RH	–	SB
22	07-10-13	ISSUED AS-BUILT PER EC 246977; INCORP. CRN 023 AND 246978; INCORP. CRN-002 (ITOP'S 13-07-008 & 017).	RV	BB	–	PMB
21	7/30/12	ISSUED AS-BUILT PER EC 246977 (PC/M 09-083)	RV	AFG	–	PMB

REV	DATE	REVISION	BY	CH	APP	APP
20	5-24-07	ISSUED AS-BUILT PER CRN-E-17461 (PC/M 07-021).	RV	BB	–	LG
19	12-8-06	ISSUED AS-BUILT PER CRN-I-4500 (PC/M 06-073).	RV	RH	TBH	TS
18	2-12-04	ISSUED AS-BUILT PER CRN-E-16500 (PC/M 03-050).	RV	RH	TBH	TS
17	7-28-94	ISSUED AS-BUILT FOR DCR-TPI-93-527 REV. 0	RV	RAS	DBJ	JCW
16	5-26-94	ISSUED AS-BUILT FOR PC/M 94-020 AND INCORP. CRN-I-1166.	RV	RH	JJN	JCW
15	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	MBP	RAS	JCW

SAFETY RELATED

POD

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

LOGIC DIAGRAM

PRESSURIZER HEATER LOGIC

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-L1

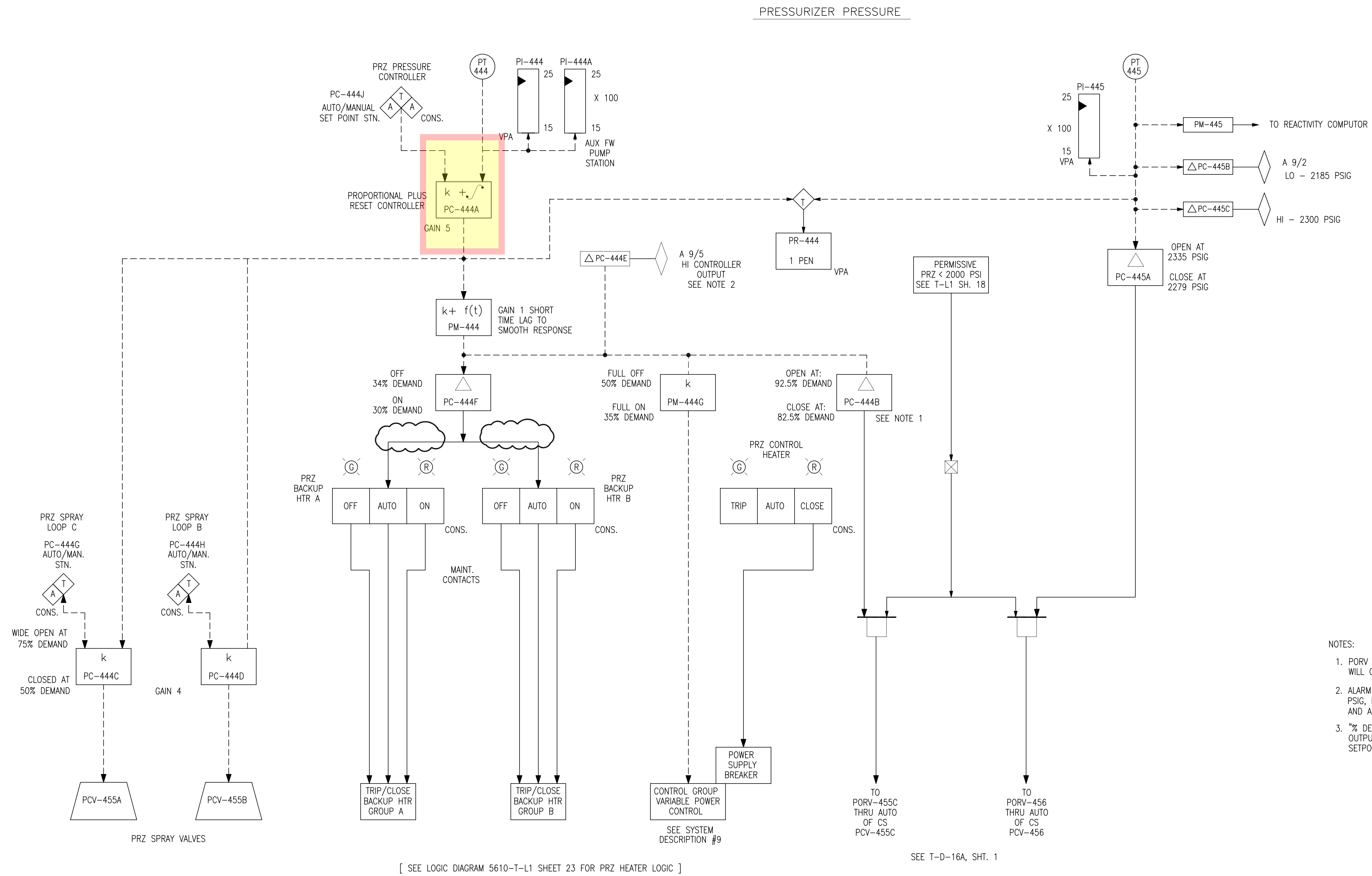
SYS –

REV 23

SHEET 23

5610-T-L1_SH 23.DWG

PRESSURIZER PRESSURE



UNIT 3 & 4
SET POINTS PRZ PRESSURE - PSIG

2465	SAFETY VALVES OPEN
2385	REACTOR TRIP
2335	PORV'S OPEN
2300	SPRAYS WIDE OPEN
	HIGH PRESSURE ALARM
2250	SPRAYS START TO OPEN
2250	CONTROL HEATERS FULL OFF
2235	NORMAL OPERATION
2220	CONTROL GP FULL ON
>2210	BACKUP HEATERS OFF (REF. ONLY)
2210	BACKUP HEATERS ON
2185	LO PRESSURE ALARM
2000	PERMISSIVE TO OPEN PORV'S
1835	REACTOR TRIP
1730	SAFETY INJECTION

NOTES:

1. PORV 456 WILL OPEN AT 2335 PSIG, PORV 455C WILL OPEN AT 92.5% DEMAND OUTPUT OF PC-444A.
2. ALARM SOUNDS AT 79% DEMAND OUTPUT OF PC-444A OR ABOUT 2300 PSIG, HOWEVER PC-444A IS A PROPORTIONAL PLUS RESET CONTROLLER AND ALARM MAY SOUND PRIOR TO 2300 PSIG.
3. "% DEMAND" REFERS TO CONTROLLER PC-444A OUTPUT AS DISPLAYED ON THE AUTO/MANUAL SETPOINT STATION.

SAFETY RELATED | POD

REV	DATE	REVISION	BY	CH	APP	APP
18	08-01-13	ISSUED AS-BUILT PER CRN-013 (EC 246978) (ITOP 13-08-002).	RV	RH	–	PMB
17	07-22-13	ISSUED AS-BUILT PER EC 246978 AND INCORP. CRN-012 (ITOP 13-07-008).	RV	RH	–	PMB
16	1-5-13	ISSUED AS-BUILT PER CRN-019 (EC 246977) ITOP 13-01-013.	AFG	BB	–	PMB
15	8-14-12	ISSUED AS-BUILT PER EC 246924 (PC/M 08-175). (PARTIAL)	RV	BB	–	JD
14	7/30/12	ISSUED AS-BUILT PER EC 246977 (PC/M 09-083).	RV	BB	–	PMB
13	7-1-12	ISSUED AS-BUILT PER EC 246924 (PARTIAL).	AFG	SB	–	BP
12	12-8-06	ISSUED AS-BUILT PER CRN-I-4500 (PC/M 06-073).	RV	BB	–	TS
11	8-5-97	ISSUED AS-BUILT PER CRN-I-3578(PC/M 97-019)	RV	RH	BP	TS
10	5-18-95	INCORPORATED CRN-I-3176 FOR PC/M 95-042.	RV	RH	JJN	JCW
9	3-8-94	INCORPORATED CRN-I-1142 FOR PC/M 93-221.	RV	RH	RAS	JCW
8	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	MSP	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4
CONTROL SYSTEM DIAGRAM
PRESSURIZER PRESSURE CONTROL

FLORIDA POWER & LIGHT
DRAWING NUMBER 5610-T-D-16B
SYS 041
REV 18
SHEET: 1

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.2	**Subsequent Actions (continued)**	
5.	**CHECK** Pressurizer Pressure stabilizing and trending to 2235 psig.	**PERFORM** the following: **A. ENSURE** proper operation of: • Pressurizer PORVs/Safeties • Pressurizer Sprays • Pressurizer Heaters **B.** IF automatic control is **NOT** effective, THEN **PLACE** controller in manual to restore Pressurizer pressure.
6.	**CHECK** following for proper operation: • Steam Gen Feed Pump Recirc • Condensate Pump Recirc • Heater Drain Pumps • Heater Drain Tank Level Controls • Secondary Heater Level Controls	

Original for Question # 6.

QUESTION 45

Given the following conditions:

- Unit 4 is at 80% power.
- 4B Steam Generator Feed Pump's breaker trips.

Subsequently

- PC-4-444J, Pressurizer Pressure Controller, does not respond in automatic.

Which ONE of the following completes the sentences below?

PC-4-444J's output is required to be initially ___(1)____ in accordance with 4-ONOP-041.5 Pressurizer Pressure Control Malfunction. PORV ___(2)___ may be operated by controller PC-4-444J.

A. (1) raised
(2) PCV-4-455C

B. (1) raised
(2) PCV-4-456

C. (1) lowered
(2) PCV-4-456

D. (1) lowered
(2) PCV-4-455C

Question # 7

Given the following sequence of events:

- ANN C 1/1, SG A NARROW RANGE LO/LO-LO LEVEL, is lit.
- ANN D 7/6, AMSAC TROUBLE/ ACTUATED, is lit.
- The RO attempts to MANUALLY trip Unit 4.
- Unit 4 remains at 100% power.

Which one of the following completes the statements below?

The crew will borate the RCS IAW __(1)__.

If neither Unit 4 Boric Acid Pumps starts, the crew will realign charging pump suction path by__(2)__.

A. (1) 4-EOP-FR-S.1, Response To Nuclear Generation/ATWS
(2) OPENING LCV-4-115B, RWST Outlet Valve, and isolating instrument air to it

B. (1) 4-EOP-FR-S.1, Response To Nuclear Generation/ATWS
(2) CLOSING LCV-4-115C, VCT Outlet Valve, and opening its breaker

C. (1) 4-EOP-E-0, Reactor Trip Or Safety Injection
(2) OPENING LCV-4-115B, RWST Outlet Valve, and isolating instrument air to it

D. (1) 4-EOP-E-0, Reactor Trip Or Safety Injection
(2) CLOSING LCV-4-115C, VCT Outlet Valve, and opening its breaker

Answer Analysis

Discussion: Without Boric Acid pumps emergency boration flowpaths through the emergency boration valve MOV-4-350 or the manual valves FCV-4-113A and B will be unsuccessful. Remaining options are through the VCT to the charging pumps or by initiating safety injection. Closing the VCT outlet valve opens the RWST flowpath to the charging pumps through 4-115B. It is necessary to open the VCT outlet MOV to prevent re-opening or opening when VCT level > 11%

A. Incorrect.
   1) Correct
   2) Plausible because the RWST outlet valve needs to be opened but this is NOT performed by failing the AOV air supply.

B. Correct

C. Incorrect.
   1) Incorrect, realignment of charging pump suction is NOT performed in E-0. Plausible because E-0 does actuate Safety Injection which will inject RWST borated water if RCS pressure drops below HHSI pump shutoff head .
   2) Plausible, see A2

D. 1) See C2
   2) Plausible , see A2

Tier: **1** **Group:** 1

K/A: 0029EK2.06; **Following an ATWS**, **Knowledge of the interrelations between** breakers, relays, and disconnects

Importance Rating: **2.9**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: **RO level question (Procedure entry criteria / purpose and overall strategy).** The K/A is matched as the applicant must evaluate the loss of a Boric Acid pumps and use the ATWS procedure FRS.1 to preserve the interrelationship between the VCT outlet valve, VCT level, (logic relay) and the RWST outlet valve (AOV). The applicant requires knowledge of MOVs, i.e. they need power to change position.

SRO Justification: N/A

Technical References: 4-EOP-FR-S.1. PTN 6910113 CVCS

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Proposed references to be provided: None

Learning Objective: **Lesson Plan** 6902346, RESPONSE TO NUCLEAR POWER GENERATION/ATWS

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

FOLDOUT Page shall be monitored for the remainder of this procedure.

3.	**Check AFW Pumps – ALL RUNNING**	Manually open steam supply valves.
4.	**Initiate Emergency Boration Of RCS:**	
	a. Verify SI – RESET	
	b. Verify Charging Pumps – AT LEAST <u>ONE</u> RUNNING IN MANUAL	
	c. Stop Makeup System	
	d. Manually start Boric Acid Pump 4A or 4B	**d.** Align Charging Pump suction to the RWST as follows: **1)** Hold closed LCV-4-115C Control switch. **2)** Direct an operator to open Breaker 40669 for LCV-4-115C. **3)** <u>WHEN</u> 40669 is open, <u>THEN</u> release LCV-4-115C Control switch. **4)** Go to Step 4.f.

Question # 8

Given the following:

- Unit 3 is at 100% power.
- Chemistry reports increased secondary activity on the 3A SG.

Subsequently:

- The control room staff has entered 3-ONOP-071.2, STEAM GENERATOR TUBE LEAKAGE.
- Performance of 3-OSP-041.1, RCS LEAK RATE CALCULATION determined the RCS leak rate to be 0.05 GPM.

Which one of the following completes the statements below?

Leakage must be classified as __(1)__ leakage.

The crew __(2)__ required to enter an action statement for LCO 3.4.6.2, Reactor Coolant System Operational Leakage.

A. (1) PRESSURE BOUNDARY LEAKAGE
(2) is

B. (1) PRESSURE BOUNDARY LEAKAGE
(2) is NOT

C. (1) IDENTIFIED LEAKAGE
(2) is

D. (1) IDENTIFIED LEAKAGE
(2) is NOT

Answer Analysis

Since the leakage is less than 150 GPD primary to secondary leakage requirement of TS 3.4.6.2, this will not require a tech spec entry since the calculated leakage is approximately 72 GPD. The leakage is also well below the 10 gpm IDENTIFIED Leakage entry condition.

A. (1) Incorrect, Plausible if the candidate confuses the definition of IDENTIFIED LEAKAGE applicability to Primary to Secondary leakage with the required Pri-Sec leakage to apply TS 3.4.6.2
(2) Incorrect, Plausible since TS 3.4.6.2 does include SG Tube Leakage > 150 GPD.

B. (1) Incorrect, See A(1)
(2) Correct

C. (1) Correct
(2) Incorrect, See A(2)

D. Correct, see discussion above

Question Number: 8

Tier: 1 **Group:** 1

K/A: 000038 (EPE 38) Steam Generator Tube Rupture

038G2.2.40; Ability to apply Technical Specifications for a system.

Importance Rating: **3.4**

10 CFR Part 55: 41.10

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: TS 3.4.6.2 and TS Definitions.

Proposed references to be provided: None

Learning Objective: PTN 6902107 OBJ. 8

Cognitive Level: Higher X

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE - 133

1.13 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GAS DECAY TANK SYSTEM

1.15 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary-to-secondary leakage).

INSERVICE TESTING PROGRAM

1.16A The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATING

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator (SG),

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4-1 up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than allowed by 3.4.6.2.e above operation may continue provided:

1. Within 4 hours verify that at least two valves in each high pressure line having a non-functional valve are in, and remain in that mode corresponding to the isolated condition, i.e., manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized. Follow applicable ACTION statement for the affected system, and

---

* Test pressure less than 2235 psig are allowed. Minimum differential test pressure shall not be less than 150 psid. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

Question # 9

Given the following:

- Unit 3 is at 100% power.

Subsequently:

- 3A SG pressure is lowering.
- RCS pressure is at 2175 psig and lowering.
- ANN C 8/2, MAIN STEAMLINE ISOLATION, is lit.
- ANN C 8/4, SG A STEAMLINE HI DP SI, is NOT lit.
- Attempts to close the 3A MSIV valve from the control room were UNSUCCESSFUL.

Which one of the following completes the statements below?

The applicable emergency procedure strategy will direct the Operators to locally CLOSE __(1)__ in order to __(2)__.

A. (1) 3A MSIV and steam trap valves
   (2) minimize the mass and energy release to containment

B. (1) 3A MSIV and steam trap valves
   (2) prevent an excessive cooldown

C. (1) 3A MSIV and Bypass Valve
   (2) minimize the mass and energy release to containment

D. (1) 3A MSIV and Bypass Valve
   (2) prevent an excessive cooldown

Answer Analysis

Discussion: Intent of question is to determine if applicant has the knowledge and comprehension to understand the purpose and mitigating strategy to address a steam leak outside containment with a failure of the associated MSIV and MSIV bypass valves to close.

Stem is required to identify the location of the leak as outside containment, not upstream of the MSIVs Both part 1 and part 2 answers depend on whether leak is inside or outside containment.

A 1) Plausible as 3A MSIV needs to be closed. Incorrect as steam traps are also closed only if leak is in containment.
2) Plausible if leak was in containment. Minimize the mass and energy release to containment would be a concern but leak was not in containment.

B 1) Plausible as 3A MSIV needs to be closed. Incorrect as steam traps are also closed only if leak is in containment.
2) Correct

C 1) Plausible Same as B1
2) Incorrect. Same as A2

D Correct.

Tier: 1 **Group:** 1

K/A: 0040AK2.01; **Knowledge of the interrelations between a Steam Line Rupture and:** Associated valves

Importance Rating: **2.6**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: **RO level question (Purpose and overall strategy).** The K/A is matched as the applicant must evaluate the stem to determine where the steam break is located and then determine which valves are needed to be isolated to mitigate the accident per procedure EOP E-2.

SRO Justification: N/A

Technical References: **E-2, Faulted Steam Generator Isolation** and bases.

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UFSAR sections

Proposed references to be provided: None

Learning Objective: PTN 6902334 OBJ. 4

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.0 OPERATOR ACTIONS

CAUTION

- At least one S/G must be maintained available for RCS cooldown.
- Any faulted S/G or secondary break is required to be maintained isolated during subsequent recovery actions unless needed for RCS cooldown.

NOTE

Foldout Page shall be monitored throughout this procedure.

1. Check Main Steamline Isolation

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Main Steamline Isolation and Bypass valves – ANY OPEN	**a.** Go to Step 2
b. Check if either Main Steam Isolation Signal has actuated: * High Steam Flow with either Low S/G Pressure 614 psig, OR Low T_{ave} 543°F OR * Hi-Hi Containment Pressure 20 psig	**b.** Push Main Steamline Isolation pushbutton on VPB, or manually close MSIV and bypass valve, on faulted SG(s). Go to Step 2
c. Verify All Main Steam Isolation and Bypass valves – CLOSED	**c.** Push Main Streamline Isolation pushbuttons on VPB or manually close valves.

BASIS DOCUMENT

PWROG Guideline Step 4 **PTN Procedure Step** 4

Isolate Faulted S/G(s)

BASIS:

Isolation of the feedwater to the faulted S/G maximizes the cooldown capability of the nonfaulted loops following a feedline break and minimizes RCS cooldown and mass and energy release following a steamline break. Isolation of steam paths from the faulted S/G also minimizes RCS cooldown and mass and energy release to containment. In addition, isolation of these steam paths could isolate the break.

When this step is reached, the operator has already performed manual actions from the Control Room to try and ensure the MSIV and bypass valve on the faulted SG(s) are closed. If any such MSIV or bypass valve is still open, local action will be required. If the SG fault is inside containment, a potential containment bypass flowpath may exist through main steam line drain and sample paths outside containment, upstream of the faulted SG MSIV. Local action should be taken to isolate these potential containment bypass flowpaths.

In addition, for the case of a SG fault inside containment with a failure of the faulted SG MSIV, or bypass valve to close, and all MSIVs and bypass valves on intact SGs closed, a potential containment bypass flowpath may exist through the various steam loads, and drain paths downstream of the MSIVs. If the faulted SG MSIV/bypass valve cannot be closed by local action, alternate local isolation should be performed to isolate the containment bypass flowpaths. If any MSIV or bypass valve on intact SGs is open, the intact SG pressure is seating the non-return valve on the faulted SG steam line, and condenser vacuum may be maintained which would be beneficial should a subsequent SGTR occur. If the MSIVs and bypass valves on intact SGs are closed, the ability to dump steam to condenser is lost, and condenser vacuum should be broken to minimize the potential for release from containment through the main steam lines.

If the following plant conditions exist:

- SG fault has occurred outside containment, but upstream of the faulted SG MSIV,
- The faulted SG MSIV/bypass valve has failed to close by automatic, or manual action,
- And any intact SG MSIV/bypass valve is open,

A containment bypass condition does not exist, however, since the intact SG pressure boundaries are relying on the non-return check valve on the faulted SG steamline, the faulted SG MSIV/bypass valve should be locally closed to preclude loss of the intact SG pressure boundary through the faulted SG non-return valve should plant conditions change to cause the faulted SG non-return valve to unseat. If local operator actions to isolate secondary piping are initiated, they should not prevent the timely termination of safety injection when the SI termination criteria are met.

Question # 10

Given the following:

- Unit 3 is at 35% power.
- The 3B SGFP is tagged out.
- The 3A SGFP is operating and maintaining SG levels.
- Feed Reg Valves are in AUTOMATIC.

Subsequently:

- 3A SGFP trips.
- MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS, fails to OPEN and the Unit supervisor directs the valve to be opened LOCALLY.

With MOV-3-1405 closed, the AFW Train 1 Flow Control Valves will be __(1)__.

When MOV-3-1405 has been LOCALLY opened, each AFW Train 1 FCV's will control flow at __(2)__.

A. (1) CLOSED
(2) 270 GPM

B. (1) CLOSED
(2) 135 GPM

C. (1) OPEN
(2) 135 GPM

D. (1) OPEN
(2) 270 GPM

Answer Analysis

A. (1) Incorrect, plausible since the candidate might believe that, the flow control valves for each train receive their open signal from their associated steam admission valves.
(2) Incorrect, the HIC for each FCV is set to 135 gpm. Plausible since each SG normally gets 270 GPM flow from a full AFW actuation (both trains).

B. (1) Incorrect.
(2) Correct. The SGFCVs get their open signal from the open signal of the steam admission valve and the HIC is set for each FVC to 135 gpm.

C. Correct, the AFW FCVs will travel to the open position when any steam supply MOV opens. In auto they are NORMALLY set to control 135 gpm.

D. (1) Correct.
(2) Incorrect, see A(2)

Question Number: 10

Tier: 1 **Group:** 1

K/A: 000054 (APE 54; CE E06) Loss of Main Feedwater
054AA1.02; Ability to operate and / or monitor the following as they apply to
the Loss of Main Feedwater (MFW):
Manual startup of electric and steam-driven AFW pumps

Importance Rating: **4.4**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-NOP-075. 5610-T-L1 SH15

Proposed references to be provided: None

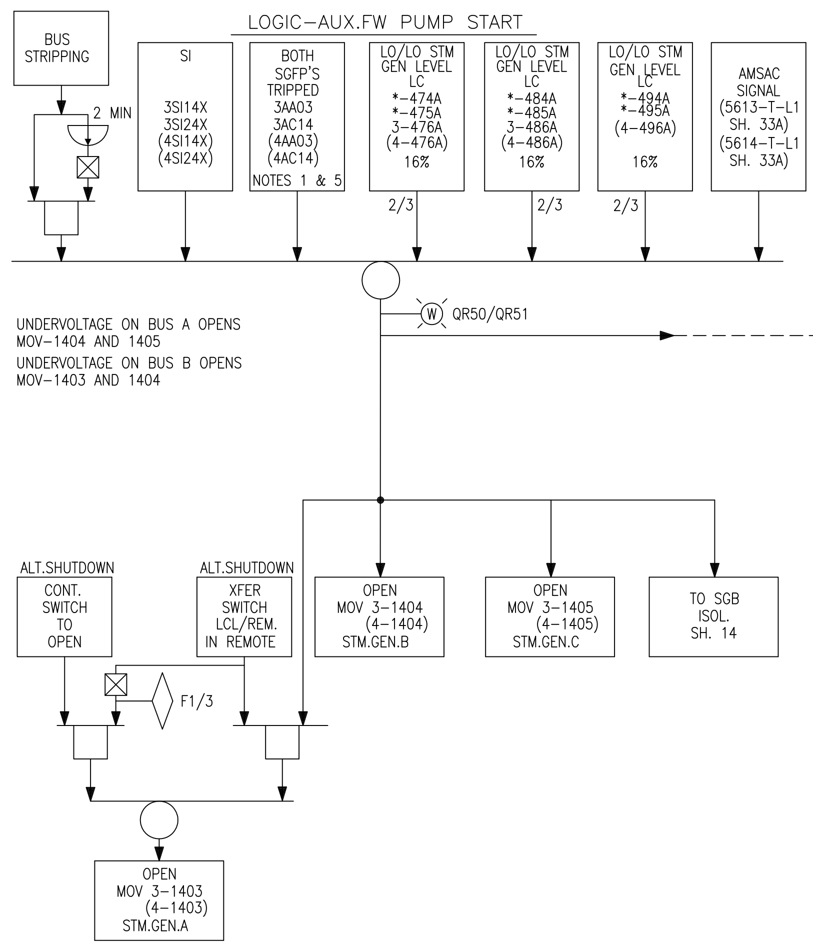
Learning Objective: **6900123 Objective 5**

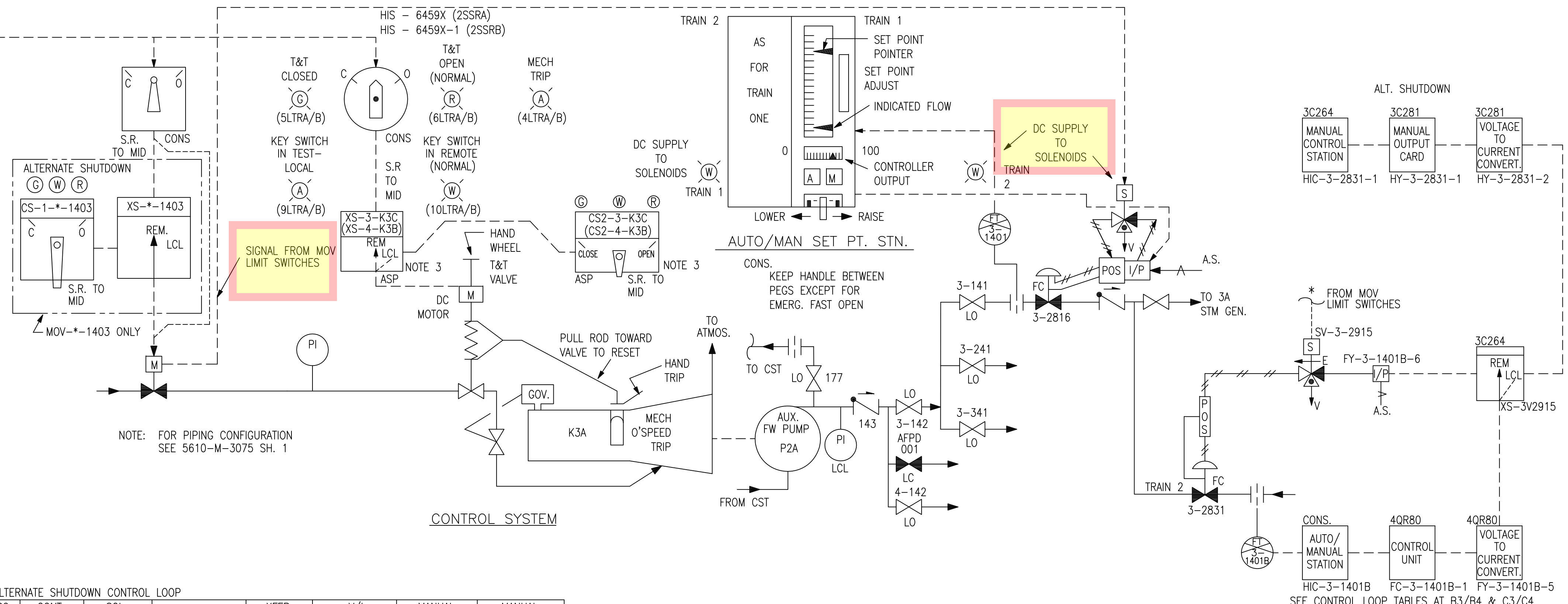
Cognitive Level: Higher _
Lower _X

Question Source: New X_
Modified Bank _
Bank _

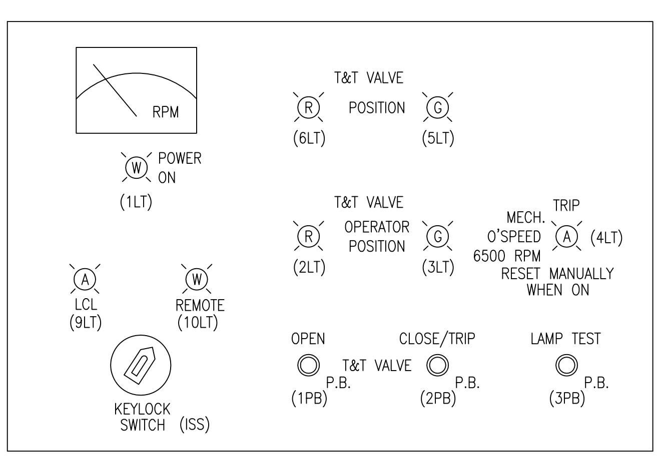
Question History: New

Comments: Screenshot of development references as applicable





LOCAL CONTROL PANEL – AUXILIARY FEEDWATER PUMP



ALTERNATE SHUTDOWN CONTROL LOOP

SG. LOOP	CONT. VALVE	SOL. VALVE	I/P	XFER. SWITCH	V/I CONVERTER	MANUAL CARD	MANUAL STATION
3A	CV–3–2831	SV–3–2915	FY–3–1401B–6	XS–3V2915	FY–3–2831–2	HY–3–2831–1	HIC–3–2831–1
3B	CV–3–2832	SV–3–2917	FY–3–1457B–6	XS–3V2915	FY–3–2832–2	HY–3–2832–1	HIC–3–2832–1
3C	CV–3–2833	SV–3–2919	FY–3–1458B–6	XS–3V2915	FY–3–2833–2	HY–3–2833–1	HIC–3–2833–1
4A							
4B							
4C							

MAIN CONTROL ROOM CONTROL LOOP

SG. LOOP	CONT. VALVE	SOL. VALVE	I/P	XFER. SWITCH	V/I CONVERTER	CONTROL UNIT	AUTO/MAN STATION
3A	CV–3–2831	SV–3–2915	FY–3–1401B–6	XS–3V2915	FY–3–1401B–5	FC–3–1401B–1	HIC–3–1401B
3B	CV–3–2832	SV–3–2917	FY–3–1457B–6	XS–3V2915	FY–3–1457B–5	FC–3–1457B–1	HIC–3–1457B
3C	CV–3–2833	SV–3–2919	FY–3–1458B–6	XS–3V2915	FY–3–1458B–5	FC–3–1458B–1	HIC–3–1458B
4A							
4B							
4C							

REFERENCE DRAWINGS:

5610–M–3075 SH. 1	AUXILIARY FEEDWATER SYSTEM TURBINE DRIVE FOR AFW PUMPS.
5610–M–3075 SH. 2	AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER PUMPS.
5610–M–1	P&ID STEAM SYSTEM.
5610–M–2	P&ID CONDENSATE & FEED–WATER SYSTEMS.

NOTES:

1. WITH ONE SGFP OFF TRIP OF THE RUNNING SGFP INITIATES AUX. F.W. PUMP START.
2. APPLICABLE TO MOV–4–1403 ONLY.
3. APPLICABLE TO AFW PUMPS "B" & "C" T&T VALVES ONLY.
4. SEE DRAWING 5610–T–L1, SH. 15A FOR AFW T & T VALVE MOTOR OPERATOR LOGIC.
5. EITHER SGFP MOTOR CONTROL SWITCH ON THE CONTROL ROOM CONSOLE MUST HAVE A RED FLAG SHOWING FOR AUTOMATIC AFW INITIATION.
6. TRAIN "A" AFW START WILL PROVIDE AN OPEN SIGNAL TO MOV–1404, MOV–1405, AND T&T VALVES MOV–6459A AND MOV–6459C.
7. TRAIN "B" AFW START WILL PROVIDE AN OPEN SIGNAL TO MOV–1403, MOV–1404, AND T&T VALVES MOV–6459B AND MOV–6459C.
8. THE OPENING OF ANY STEAM ADMISSION MOV WILL PROVIDE AN OPEN SIGNAL TO ALL SIX AUXILIARY FEEDWATER FLOW CONTROL VALVES ON THAT UNIT.

SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP
29	3–10–03	ISSUED AS–BUILT PER PC/M 01–012.	RH	RRW	TEB	BAP
28	4–04–02	ISSUED AS–BUILT PER PC/M 01–059.	RH	RRW	TEB	TS
27	3–23–99	ISSUED AS–BUILT PER PC/M 97–033.	RV	RH	MCP	RB
26	10–17–98	ISSUED AS–BUILT PER PC/M 97–033. (PARTIAL)	RV	RH	AD	BT
36	5–21–15	ISSUED AS–BUILT PER EC 283083.	RH	AFG	–	BB

REV	DATE	REVISION	BY	CH	APP	APP
35	12–18–13	ISSUED AS–BUILT PER EC–DCR 280474.	JJW	BB	–	MS
34	05–01–13	ISSUED AS–BUILT PER EC 247049 (ITOP 13–04–155).	RV	BB	–	PMB
33	7–17–12	ISSUED AS–BUILT PER EC 247048 (PC/M 10–039).	RV	BB	–	PMB
32	6–8–10	ISSUED AS–BUILT PER CRN E–18200 (PC/M 10–058).	RH	BB	–	PDS
31	3–31–10	ISSUED AS–BUILT PER CRN–E–18036 (PC/M 09–106).	RV	BB	–	LG
30	3–3–08	ISSUED AS–BUILT PER PER CRN–I–4723 (PC/M 07–095).	RV	RH	–	LG

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

LOGIC DIAGRAM

AUXILIARY FEEDWATER PUMP START LOGIC, P&I AND CONTROL SYSTEM

POD

DRAWING NUMBER 5610–T–L1

SHEET 15

SYS 075

REV 36

85610–T–L1_SH 15.DWG

4.3 Shutdown (continued)

NOTE

In MODES 1, 2, 3, or 4, the Hand Indicating Controllers (HIC) for the Auxiliary Feedwater Regulating Valves are required to be energized and in AUTO mode of operation preset to 135 gpm.

Concurrent Verification

10. **ADJUST** the following controllers setpoint to 135 gpm and **PLACE** controller in AUTO:

- HIC-3-1401B, TRAIN 2 AFW FLOW TO 3A S/G ______ ______ CV
- HIC-3-1457B, TRAIN 2 AFW FLOW TO 3B S/G ______ ______ CV
- HIC-3-1458B, TRAIN 2 AFW FLOW TO 3C S/G ______ ______ CV

11. **REDUCE** feed flow on each of the following controllers to 0 gpm:

- HIC-3-1401A, TRAIN 1 AFW FLOW TO 3A S/G (CV-3-2816)
- HIC-3-1457A, TRAIN 1 AFW FLOW TO 3B S/G (CV-3-2817)
- HIC-3-1458A, TRAIN 1 AFW FLOW TO 3C S/G (CV-3-2818)

12. **CLOSE** MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS. ______ ______ IV

Question # 11

Given the following:

- Both units experience a loss of all AC.
- Neither unit's A nor B 4kV buses can be PROMPTLY re-energized from the control room.
- 3/4-EOP-ECA-0.0, Loss of all AC power, are in progress.

Which one of the following completes the statements below?

IAW 3/4-EOP-ECA-0.0, the crew is required to shed Vital DC loads within a MAXIMUM of ______ minutes.

A. 30

B. 60

C. 90

D. 120

Answer Analysis

A. Incorrect, plausible since 30 minutes is associated with the 3-EOP-E-0 time limit for placing PAHMS in service.

B. Incorrect, plausible since 60 minutes is the 3-EOP-ECA-0.0 time limit for starting Computer Room Chiller.

C. Correct. IAW 3-EOP-ECA-0.0

D. Incorrect, plausible since 120 minutes is the design limit for the Vital DC Batteries. Part 2 correct.

Tier: 1 **Group:** 1

K/A: 0029EK2.06; **Knowledge of the operational implications of:** battery discharge rates, **as they apply to a Station Blackout**

Importance Rating: **3.3**

10 CFR Part 55: 41.8

10 CFR 55.43.b **:** N/A

K/A Match: **RO question. Addresses Purpose and Overall strategy to prolong battery capacity until Flex procedures and actions can be installed.**

SRO Justification: N/A

Technical References: 3-EOP-ECA-0.0

Proposed references to be provided: None

Learning Objective: Lesson Plan PTN6910139; Malfunctions in the Vital DC and or AC system (Objective 11)

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

CAUTION

If both Units have experienced a Loss Of All AC Power, AND **neither** Unit's A **NOR** B 4KV Buses can be promptly re-energized, then Vital DC Load Shedding per 0-FSG-04 is required to be completed within 90 minutes of event initiation.

NOTE

An Extended Loss of AC Power (ELAP) exists if either of the following occurred:

* Dual unit Loss Of All AC Power

  OR

* Single unit Loss Of All AC Power with inability to meet SBO power restoration time requirements

9. Check If ELAP In Progress

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check opposite unit 4KV buses (A AND B) – AT LEAST ONE ENERGIZED	**a.** Go to Step 9.d.
b. Check either of the following – * Elapsed time since reaching Attachment 6, Step 5 NOTE – GREATER THAN 10 MINUTES OR * Elapsed time since reaching Attachment 7, Step 4 NOTE – GREATER THAN 10 MINUTES	**b.** WHEN greater than 10 minutes has elapsed, THEN continue with Step 9.v.
c. Go to Step 9.v	
d. Initiate Containment Isolation Phase A	

9.0 GRAPHIC DISPLAY/READOUT EQUIPMENT

General purpose displays located in the control room, the computer room, the Technical Support Center and the Emergency Operating Facility, provide access to a comprehensive set of system-oriented mimic, tabular and trend displays. In addition specific user-defined display capability is available. The DCS (ERDADS) and the DCS (SPDS) graphics are displayed on workstation and flat panel display. A continuous calorimetric flat panel display is also provided in the control room. A DCS (SPDS) touch screen display is used as a SPDS graphics navigation tool. The DCS operator workstations display ERDADS information in the Control Room. QSPDS display 3B is mounted in the Operator Console (3C256). C26

10.0 SYSTEM POWER

Power is supplied to the DCS/SAS/ERDADS through four Uninterruptible Power Systems (UPS), which provide redundant supplies to the computer. Each UPS is comprised of a distribution panel, a static transfer switch, a static inverter, and a regulating transformer. The UPS ensures that the DCS/SAS/ERDADS computer installation is not operationally impaired by power fluctuations or failures. C26 C26

Single phase, 120 volt, 60 Hz power is supplied to the distribution panels through primary or alternate feeds by means of static transfer switches. The primary feeds are two - 20 kVA and two - 10 kVA static inverters that are powered by the auxiliary power upgrade 125V DC buses. In the event of a failure in the primary feed, static transfer to the alternate feed is accomplished. Manual switching is provided for maintenance purposes. The alternate feeds are from the Condensate Polishing System Motor Control Center via 480/120 volt, single phase, regulating transformers (two - 10 kVA and two - 25 kVA). These power supplies are non-safety related.

The DCS ERDADS and PDN systems are powered from sources that collectively provide a robust power supply capable of withstanding a short duration (<2hours) Loss of Offsite Power (LOOP) coincident with loss of any single panel, inverter, battery, or AC power feed without the interruption of service. This is accomplished by employing NNS station inverters for powering DCS and PDN equipment. Critical components are provided with redundant power feeds and are designed to automatically switch to the redundant source upon loss of one of the power sources. One of the redundant feeds to each of the PDN Zone switches in the Control Room is provided from a Safety Related inverter to provide PDN functionality during an extended LOOP. Extended LOOP is that period of time after which the NNS batteries have discharged (approximately 2 hours). This will provide additional DCS (ERDADS) functional capability by having one safety related source powering the PDN Zone switches. C26

ATTACHMENT 5
Unit 3 Component KW Load Rating Chart
(Page 1 of 2)

CAUTION

- Steady state loading on each Unit 3 EDG shall **NOT** exceed 2500 KW. Load transients up to 2750 KW are acceptable when starting additional equipment.
- When using this chart for Attachment 4 with only one opposite unit bus energized, and with one opposite unit EDG supplying power via the SBO Tie, the 2000-hour rating in parentheses may be used.
- Steady state loading on each opposite unit EDG shall **NOT** exceed 2874(3095) KW. Load transients up to 3162(3195) KW are acceptable when starting additional equipment.

NOTE

- If power is available, one Computer Room Chiller is required to be restarted within 60 minutes of a Loss of Offsite Power to maintain operability of DCS and QSPDS.
- Battery Charger load is dependent on the status of its parallel charger (i.e., in service or de-energized).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- Foldout Page shall be monitored for the remainder of this procedure.
- Hydrogen Monitors should be in service within 30 minutes of a valid SI signal.

5. Continue With Attachment 3 To Complete The Prompt Action Verifications While Performing This Procedure

6. Check AFW Pumps – AT LEAST TWO RUNNING

Perform the following:

a. Manually open valves to establish two AFW Pumps running.

b. IF an AFW Pump is tripped, THEN dispatch an operator to locally reset the AFW turbine trips.

c. IF both units require AFW, THEN perform the following:

1) Verify all RCPs are tripped.

2) Establish 340 gpm flow to each unit.

3) Use a setpoint of 340 gpm for required AFW flow instead of the 400 gpm specified in subsequent steps and procedures.

7. Verify AFW Valve Alignment – PROPER EMERGENCY ALIGNMENT

Manually align valves to establish proper AFW alignment.

Question # 12

Given the following:

- Unit 4 experiences a LOOP while in MODE 3.
- Emergency Buses are energized with the EDGs.

Which one of the following completes the statements below?

While the EDGs are carrying the buses, manipulating the Speed Changer Switch will change EDG __(1)__ and manipulating the EDG Voltage Regulator Switch will change EDG __(2)__.

A. (1) KiloWatts
(2) Voltage

B. (1) KiloWatts
(2) VARs

C. (1) Frequency
(2) Voltage

D. (1) Frequency
(2) VARs

Answer Analysis

A. Incorrect.
   Part 1 incorrect, plausible since this is correct when the EDG is synchronized to an energized bus.
   Part 2 correct.

B. Incorrect.
   Part 1 incorrect, plausible since this is correct when the EDG is synchronized to an energized bus.
   Part 2 incorrect, plausible since this is be correct if the EDG was paralleled to the bus.

C. Correct. While the EDG is carrying the bus it will dictate frequency and voltage.

D. Incorrect.
   Part 1 correct.
   Part 2 incorrect, plausible since this is be correct if the EDG was paralleled to the bus.

Question Number: 12

Tier: 1 **Group:** 1

K/A: 000056 (APE 56) Loss of Offsite Power

056AA1.04; Ability to determine and interpret the following as they apply to

the Loss of Offsite Power:

Adjustment of speed of ED/G to maintain frequency and voltage levels

Importance Rating: **3.2**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 4-ONOP-004.1

Proposed references to be provided: None

Learning Objective: PTN 6902136 OBJ. 11

Cognitive Level: Higher X _

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C A U T I O N

When the Startup Transformer breaker is closed, the EDG load will drop to zero. To prevent potential damage to the EDG, the EDG output breaker should be opened as soon as the Startup Transformer is verified to be carrying 4KV bus load.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	**Synchronize Unit 4 Startup Transformer To 4A 4KV Bus**	
	a. Place Startup Transformer Sync to 4A 4KV Bus 4AA05 to ON	
	b. Verify running voltage matched incoming voltage	b. Adjust 4A Diesel Generator Voltage Regulator.
	c. Verify synchroscope - ROTATING SLOWLY IN FAST DIRECTION	c. Adjust 4A Diesel Generator Speed Changer.
	d. **WHEN** synchroscope pointer reaches 12 o'clock position, **THEN** close Startup Transformer 4A 4KV Bus Supply, 4AA05	d. **IF** Startup Transformer 4A 4KV Bus Supply, 4AA05, can **NOT** be closed, **THEN** perform the following: 1) Place Startup Transformer Sync to 4A 4KV Bus 4AA05 to OFF **AND** remove handle. 2) Direct Electrical Maintenance to determine cause of breaker failure. 3) Go to Step 35.
	e. Place Startup Transformer Sync to 4A 4KV Bus 4AA05 to OFF **AND** remove handle	

Question # 13

Given the following:

- Unit 3 is at 100% power
- N-3-43, Power range channel, is de-energized for maintenance
- 3C Charging pump is running

Subsequently:

- ANN J5/5, P08 BREAKER TRIPPED, alarms
- The 3C Charging Pump AUTO control light goes dark as follows:

ALSO PROVIDED AS A REFERENCE IN LARGER FORMAT

Which one of the following completes the statements below?

An IMMEDIATE reactor trip __(1)__ occur.
The RO __(2)__ able to control the 3C Charging Pump speed by operating its controller in MANUAL.

A. (1) will NOT
(2) is

B. (1) will NOT
(2) is NOT

C. (1) will
(2) is

D. (1) will
(2) is NOT

Answer Analysis

A. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since for a loss of P09 the 3C CHP is able to be controlled MANUALLY.

B. Correct. 3P08 is the power supply to N-3-43, only 1/4 PRNI are in the tripped condition, and no RPS actuation occurs. Upon loss of 3P08 the 3C CHP goes to AUTO Lock up and cannot be operated MANUALLY.

C. Incorrect.
Part 1 incorrect, plausible since multiple failures that affect RPS have occur.
Part 2 incorrect, plausible since for a loss of P09 the 3C CHP is able to be controlled MANUALLY.

D. Incorrect.
Part 1 incorrect, plausible since multiple failures that affect RPS have occured.
Part 2 correct.

Tier: 1 **Group:** 1

K/A: 0057AA1.02; **Loss of Vital AC Instrument Bus / 6.** Manual PZR level control

Importance Rating: **3.8**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must evaluate the conditions and annunciators provided, evaluate the expected response, and determine the operator overall action required to mitigate the event.

SRO Justification: N/A

Technical References: 3-ONOP-003.8

Proposed references to be provided: None

Learning Objective: PTN 6902260 OBJ. 4

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

ENCLOSURE 1
(Page 1 of 3)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON FAILURE OF VITAL INSTRUMENT PANEL 3P08

FUNCTIONS, OPERATING

Loss of Auto control of C Feedwater Control Valve, FCV-3-498, shifts to Backup Controller
PRMS MONITORS due to loss of power
  Lose S/G Blowdown, causes Steam Generator level to increase
  Loss of power to R-11 and R-12, initiates Control Room **AND** Containment Ventilation Isolation
  Lose Liquid/Gas Release
Steam dump to condenser valves receive trip open signal but no Arming Signal
Lose Auxiliary Feedwater Train 2 Controllers (3)
3C Charging Pump controller locks up as is
Lose automatic operation of PORV PCV-3-456 (if OMS in normal OPS)
Disarms AMSAC due to loss of PT-446 (after six minute time delay)
Loss of 3B QSPDS (If 3B Inverter and CVT lost)
Possible loss of power to hand/auto station for CV-3-1608 if aligned to 3P08 and would fail closed

INDICATORS

FI-3-110	Emerg Borate Flow
TI-3-116	VCT Temperature
PI-3-117	VCT Pressure
PI-3-156	A RCP P2 Seal Pressure
TI-3-432B	C Loop OVPWR ΔT
TI-3-432A	C Loop ΔT
TI-3-432C	C Loop OVTEMP ΔT
TI-3-432D	C Loop Temp Avg
PI-3-457	Pzr Pressure
LI-3-461	Pzr Level/Cont Ch III
FI-3-416	A Loop RCS Flow
FI-3-426	B Loop RCS Flow
FI-3-436	C Loop RCS Flow
PI-3-445	Pzr Pressure

Question # 14

Given the following:

- Unit 3 experiences an SI from 100% power.
- 3-EOP-E-0, Reactor Trip or Safety Injection, is in progress.

Which one of the following completes the statements below?

CV-3-2803, Instrument Air Containment Isolation is expected to be in the __(1)__ position.

Instrument Air supply to containment is essential for operation of __(2)__ in subsequent steps in the EOP Network.

A. (1) OPEN
(2) Pressurizer Spray valves.

B. (1) OPEN
(2) HCV-3-121, Charging Flow To Regen Heat Exchanger.

C. (1) CLOSED
(2) Pressurizer Spray valves.

D. (1) CLOSED
(2) HCV-3-121, Charging Flow To Regen Heat Exchanger.

Answer Analysis

Containment Isolation Signal does not close the Instrument Air to Containment Isolation valve at PTN. As per the Design Basis. Also, the design basis states that the verification of IA to Containment is prioritized in the procedure to ensure equipment in subsequent steps can be operated. Pressurizer spray is needed to stabilize RCS pressure when SI Termination criteria is met.

A. (1) Correct
(2) Correct

B. (1) Correct
(2) Incorrect, plausible because the HCV-121 is required to be operated when restoring Charging flow per EOP-2 Attachment 2, however this valve gets Instrument Air from the Aux Building IA Header.

C. (1) Incorrect, Plausible since this is a containment isolation valve, however, it does not get a CIV signal and fails open.
(2) Correct

D. (1) Incorrect, see C(1)
(2) Incorrect. See (B(2)

Question Number: 14

Tier: 1 **Group:** 1

K/A: 000065 (APE 65) Loss of Instrument Air

065AK3.08; Knowledge of the reasons for the following responses as they apply to

the Loss of Instrument Air:

Actions contained in EOP for loss of instrument air

Importance Rating: **3.7**

10 CFR Part 55: 41.5, 10

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 5613-M-3047 SH 2. 3-EOP-E-0.

Proposed references None

to be provided:

Learning Objective: PTN 6902145 OBJ. 10

Cognitive Level: Higher X _

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

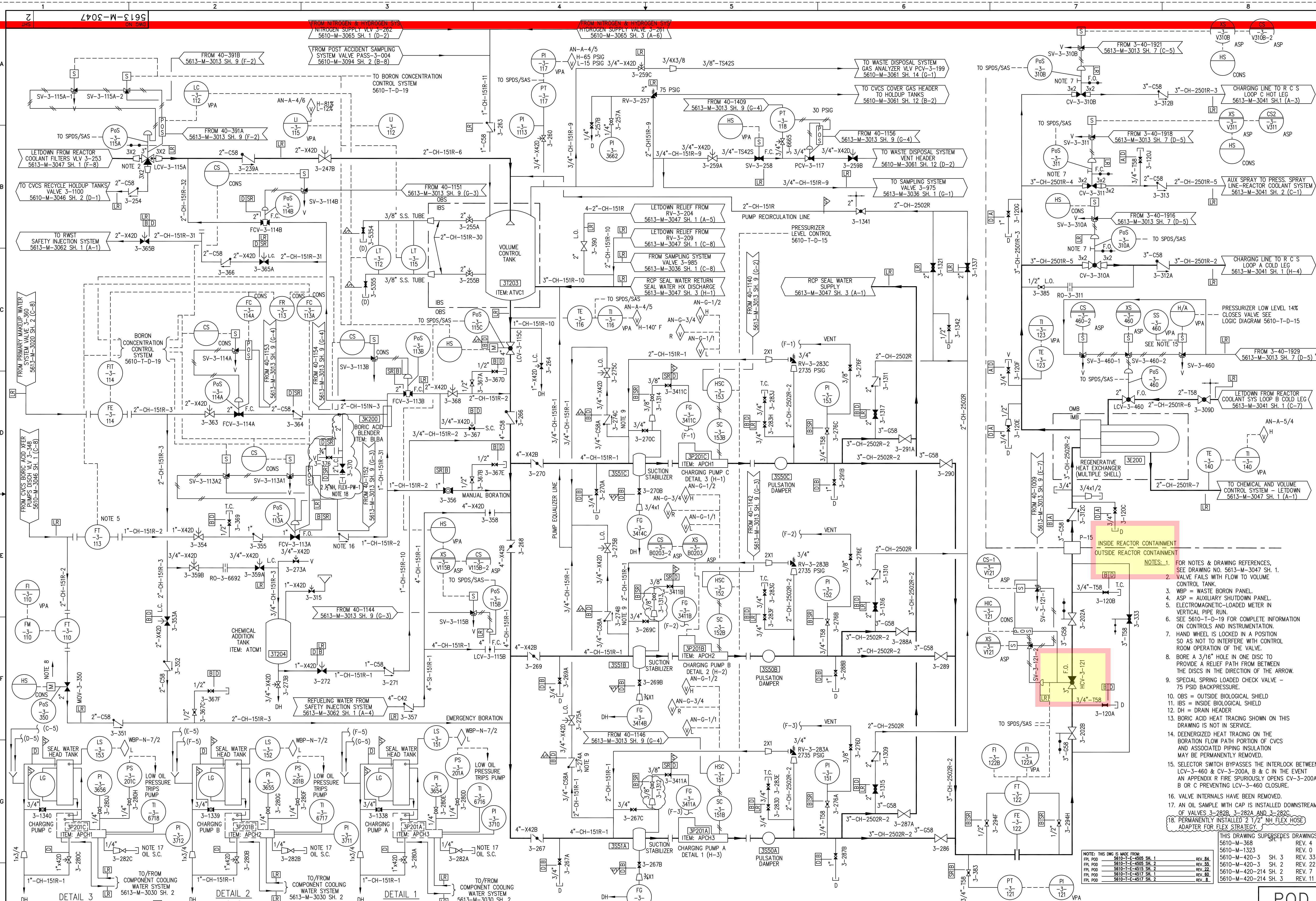
Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
24.	**Verify Instrument Air To Containment**	
	a. CV-3-2803, Instrument Air Containment Isolation – OPEN	**a.** Manually open valve.
	b. PI-3-1444, Instrument Air Pressure – GREATER THAN 95 PSIG	**b.** Restore Instrument Air pressure using 3-ONOP-013, LOSS OF INSTRUMENT AIR, <u>while</u> continuing this procedure.

<u>CAUTION</u>

If RCS pressure decreases in an uncontrolled manner to less than 275 psig, manual action will be required to restart the RHR pumps after they have been secured.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
➔ **25.**	**Check If RHR Pumps Should Be Stopped**	
	a. Check RCS pressure:	
	1) Pressure – GREATER THAN 275 PSIG	**1)** Go to 3-EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	2) Pressure – STABLE <u>OR</u> INCREASING	**2)** Go to Step 26.
	b. Stop RHR Pumps and place in standby	
26.	**Check Power Supply To All Charging Pumps – ALIGNED TO OFFSITE POWER**	Perform the following: **a.** Check diesel capacity adequate to run at least <u>one</u> Charging Pump. **b.** <u>IF</u> diesel capacity is **NOT** adequate, <u>THEN</u> shed non-essential loads. Refer to Attachment 2 for component KW load rating.



NOTES:
1. FOR NOTES & DRAWING REFERENCES, SEE DRAWING NO. 5613-M-3047 SH. 1.
2. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK.
3. WBP = WASTE BORON PANEL.
4. ASP = AUXILIARY SHUTDOWN PANEL.
5. ELECTROMAGNETIC-LOADED METER IN VERTICAL PIPE RUN.
6. SEE 5610-T-D-19 FOR COMPLETE INFORMATION ON CONTROLS AND INSTRUMENTATION.
7. HAND WHEEL IS LOCKED IN A POSITION SO AS NOT TO INTERFERE WITH CONTROL ROOM OPERATION OF THE VALVE.
8. BORE A 3/16" HOLE IN ONE DISC TO PROVIDE A RELIEF PATH FROM BETWEEN THE DISCS IN THE DIRECTION OF THE ARROW.
9. SPECIAL SPRING LOADED CHECK VALVE – 75 PSID BACKPRESSURE.
10. OBS = OUTSIDE BIOLOGICAL SHIELD
11. IBS = INSIDE BIOLOGICAL SHIELD
12. DH = DRAIN HEADER
13. BORIC ACID HEAT TRACING SHOWN ON THIS DRAWING IS NOT IN SERVICE.
14. DEENERGIZED HEAT TRACING ON THE BORATION FLOW PATH PORTION OF CVCS AND ASSOCIATED PIPING INSULATION MAY BE PERMANENTLY REMOVED.
15. SELECTOR SWITCH BYPASSES THE INTERLOCK BETWEEN LCV-3-460 & CV-3-200A, B & C IN THE EVENT AN APPENDIX R FIRE SPURIOUSLY OPENS CV-3-200A, B OR C PREVENTING LCV-3-460 CLOSURE.
16. VALVE INTERNALS HAVE BEEN REMOVED.
17. AN OIL SAMPLE WITH CAP IS INSTALLED DOWNSTREAM OF VALVES 3-282B, 3-282A AND 3-282C.
18. PERMANENTLY INSTALLED 2 1/2" NH FLEX HOSE ADAPTER FOR FLEX STRATEGY.

NOTE: THIS DWG IS MADE FROM:

FPL POD	5610-T-E-4505 SH. 1	REV. 84
FPL POD	5610-T-E-4505 SH. 2	REV. 55
FPL POD	5610-T-E-4515 SH. 2	REV. 22
FPL POD	5610-T-E-4517 SH. 1	REV. 60
FPL POD	5610-T-E-4517 SH. 2	REV. 8

THIS DRAWING SUPERSEDES DRAWINGS:

5610-M-368	SH. 1	REV. 4
5610-M-1323		REV. 0
5610-M-420-3	SH. 3	REV. 33
5610-M-420-3	SH. 2	REV. 22
5610-M-420-214	SH. 2	REV. 7
5610-M-420-214	SH. 3	REV. 11

POD

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
55	12/13/12	ISSUED AS-BUILT PER CRN M-12391 (EC 241742). (ITOP 12-12-043)	AFG	CBW	–	DAS	49	04-04-07	ISSUED AS-BUILT PER CRN M-11984 (PC/M 06-108).	RH	RV	LG	–
54	5/24/12	ISSUED AS-BUILT PER EC 270716.	RV	AFG	JG	AC	58	11-3-15	ISSUED AS-BUILT PER EC 280631.	LR	SB	–	DA
53	9/28/10	ISSUED AS-BUILT PER EC 241979.	RV	BB	BSG	PRB	57	3-14-14	ISSUED AS-BUILT PER EC 242460.	RH	CW	MG	–
52	8-20-10	ISSUED AS-BUILT PER CRN'S I-4272 & M-11882 (PC/M 04-047).	RV	BB	BSG	PRB	56	1/25/13	ISSUED AS-BUILT PER EC-DCR 277328.	AFG	BB	–	GC
51	6-14-10	ISSUED AS-BUILT PER PC/M 04-047; INCORP. CRN'S M-11882 & I-4272 (Partial)	RH	RV	BSG	PRB	0	3-5-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-045.	MAB	MD	RAS/SVM	LRB
50	4/1/10	ISSUED AS-BUILT PER CRN E-18146 (PC/M 10-007).	RH	RV	–	LG	—	—		—	—	—	—

FPL

TURKEY POINT NUCLEAR UNIT 3

P & ID

CHEMICAL AND VOLUME CONTROL SYSTEM
CHARGING AND LETDOWN

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3047

SYS 047

REV 58

SHEET 2

5613-M-3047_SH 2

Question # 15

Given the following:

- Unit 3 is at 100%
- 3A EDG testing is in progress IAW 3-OSP-023.1, Diesel Generator Operability Test.
- 3A EDG is operating in the LAG.

Which one of the following completes the statements below?

In order to raise EDG output VARS, the RO must adjust the EDG voltage regulator in the __(1)__ direction.
3-OSP-023.1, Diesel Generator Operability Test __(2)__ adjusting grid voltage while the EDG is on the bus.

A. (1) RAISE
(2) cautions against

B. (1) RAISE
(2) recommends

C. (1) LOWER
(2) cautions against

D. (1) LOWER
(2) recommends

Answer Analysis

A. Correct. While in the lag, the EDG will pick up reactive load when excitation voltage is raised. IAW 3-OSP-023.1, grid voltage adjustments should be avoided while the EDG is paralleled to the bus.

B. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since it is possible to change grid reactive while the EDG is synchronized to the bus.

C. Incorrect.
Part 1 incorrect, plausible since while in the lead, lowering excitation voltage will increase reactive load magnitude.
Part 2 correct.

D. Incorrect.
Part 1 incorrect, plausible since while in the lead, lowering excitation voltage will increase reactive load magnitude.
Part 2 incorrect, plausible since it is possible to change grid reactive while the EDG is synchronized to the bus.

Question Number: 15

Tier: 1 **Group:** 1

K/A: 000077 (APE 77) Generator Voltage and Electric Grid Disturbances

077AK1.02; Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances:

Over-excitation

Importance Rating: **3.3**

10 CFR Part 55: 41.4, 5, 7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must be familiar with GOI-301 and OSP 023.1 cautions and warnings. The applicant must also demonstrate knowledge of how a lowering of grid voltage causes increased excitation on the main and EDG field amps to attempt to raise the voltage.

SRO Justification: N/A

Technical References: 3-OSP-023.1.

Proposed references None

to be provided:

Learning Objective: PTN 6902136 OBJ. 11

Cognitive Level: Higher X_

Lower _

Question Source: New X_

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

4.0 PRECAUTIONS/LIMITATIONS

4.1 For test purposes, the generator load shall not exceed 2750 kW and generator current shall not exceed 477 amps, the basic overload rating.

4.2 Extreme caution should be used when adjusting grid voltage while the EDG is paralleled to the grid. The resulting voltage change could cause the EDG to become under or over excited resulting in a lockout of the EDG. Adjustments to grid voltage while the EDG is on the associated bus should be avoided. If it is necessary to adjust voltage during this time, do it using the guidance of Enclosure 3.

4.3 An electric motor driven soakback pump shall be in operation at all times when the engine is not running. This is necessary to provide turbocharger prestart lubrication and post shutdown bearing cooling.

4.4 Do not allow the engine to run unloaded at 900 rpm for periods in excess of 4.5 hours. The EDGs should not be operated at less than 25 percent load due to the accumulation of lube oil in the exhaust during light load operation (souping). Depending on the amount of souping that has taken place, an exhaust fire could result when the engine is suddenly loaded, raising exhaust temperatures quickly.

4.5 After 4.5 cumulative hours of operation at synchronous speed at loads between 0 and 20 percent (0-500 kW), the engine shall be run at a minimum of 40 percent load for at least 30 minutes to clean out the oil residual that accumulates in the exhaust stack.

4.6 After 8 hours of continuous operation at idle speed, the engine shall be run at a minimum of 50 percent load for at least 30 minutes to clean out the oil residual that accumulates in the exhaust stack.

4.7 During testing, only one of the Unit 3 EDGs shall be paralleled with the off-site transmission network at a time.

4.8 During testing, only one of the EDGs for each unit shall have its Master Control Switch in the Local position with the Rapid Start/Auto Start Bypass Switch positioned to Bypass. When the switches are in this configuration, the EDG will not automatically start upon loss of bus voltage or a Safety Injection signal.

4.9 When the EDGs are in Standby Mode, the Governor Control Switches and the Voltage Adjust Control Switches at the Local panels and the Diesel Gen Speed Changer switches and the Diesel Gen Volt Regulator switches at the Control Room panels shall not be operated. Actuation of these switches will alter the preset speed or voltage settings.

4.10 Technical Specification requirements shall be observed and any deviation from these requirements shall be reported immediately to the Shift Manager. Technical Specifications should be consulted for any change in system status.

4.11 Hearing protection shall be worn in the EDG Rooms when operating the Emergency Diesel Generator.

4.12 The Shift Manager shall be notified immediately if any acceptance criteria are not met or any malfunction or abnormal conditions occur. This information shall also be recorded under Remarks at the end of the applicable attachment.

INIT 7.1.2.32 (Cont'd)

NOTES

- *The following voltage adjustment will place the generator reactive load in Lag.*
- *As the generator is loaded, reactive load should be maintained in Lag as observed on the local Varmeter. The desired reactive load (vars) should be maintained approximately 300 to 1000 kVars.*
- *Coordination with the local operator is required to adjust reactive load.*

k. Using the 3A Diesel Gen Volt Regulator, perform the following to adjust reactive load:

____ (1) **IF** the local Varmeter indicates Lag, **THEN** slowly adjust voltage until the reactive load (vars) is approximately 300 to 1000 kVars in Lag.

(2) **IF** the local Varmeter indicates Lead **OR** **IF** vars are zero, **THEN** perform the following:

____ (a) Slowly Raise the voltage until the amps stop decreasing and start to increase (Varmeter indicates Lag).

____ (b) Adjust voltage until the ratio of load (watts) to reactive load (vars) is between 300 to 1000 kVars in Lag.

____ (3) Request the local Operator to periodically report Varmeter reading as generator load is increased.

____ 33. **IF** the EDG Volts/Frequency Recorder was connected in preparation for Engineered Safeguards Integrated Testing, **THEN** notify Engineering to verify proper operation.

____ 34. Inspect the 3A EDG for any leaks or abnormalities.

____ a. Inspect the bucket under the air box drain for any additional accumulation of fluids resulting from the start, and record results on Attachment 2, Section 2.

____ b. Inspect the tell-tale weep holes on each cooling water pump drive shaft housing for signs of leakage of cooling water or oil from the seal.

____ (1) **IF** leakage is detected from the cooling water pump, **THEN** notify the System Engineer.

c. **IF** cooling water leakage is found, **THEN** perform the following:

____ (1) **IF** the leakage has the potential to enter a floor drain, **THEN** plug affected floor drains.

____ (2) Notify the Shift Manager and Chemistry.

Question # 16

Given the following:

- Unit 4 experiences a Safety Injection.
- The crew enters 4-EOP-ECA-1.2, LOCA Outside Containment.

Which one of the following completes the statements below?

The crew will FIRST verify that __(1)__ are CLOSED since __(2)__.

A. (1) MOV-4-750 and MOV-4-751, RHR Suction Valves
(2) there is a higher probability of a break in this system

B. (1) MOV-4-750 and MOV-4-751, RHR Suction Valves
(2) these valves are easier to access for LOCAL isolation

C. (1) MOV-4-843A and MOV-4-843B, HHSI to Cold Legs
(2) there is a higher probability of a break in this system

D. (1) MOV-4-843A and MOV-4-843B, HHSI to Cold Legs
(2) these valves are easier to access for LOCAL isolation

Answer Analysis

A. Correct. IAW 4-EOP-ECA-1.2, step 1 verifies that the RHR Pump Suction Valves are closed.

B. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since the ECCS and RHR systems have components that are located inside and outside of the containment building.

C. Incorrect.
Part 1 incorrect, these valves would be manipulated later in 4-EOP-ECA-1.2.
Part 2 correct.

D. Incorrect.
Part 1 incorrect, these valves would be manipulated later in 4-EOP-ECA-1.2.
Part 2 incorrect, plausible since the ECCS and RHR systems have components that are located inside and outside of the containment building.

Question Number: 16

Tier: 1 **Group:** 1

K/A: (W E04) LOCA Outside Containment

WE04K3.4; Knowledge of the reasons for the following responses as they apply to the (LOCA Outside Containment) :

RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated

Importance Rating: 3.6

10 CFR Part 55: 41.5, 10

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 4-EOP-ECA-1.2. BD-EOP-ECA-1.2.

Proposed references to be provided: None

Learning Objective: PTN 6902333 OBJ. 10

Cognitive Level: Higher _

Lower _X

Question Source: New X

Modified Bank

Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

1. Verify Proper Valve Alignment:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check both Loop 4A RHR Suction Stop Valves – CLOSED: * DCS, Containment Isolation Valve Position screen: • MOV-4-750 • MOV-4-751 * Locally, at ASP: • MOV-4-750 • MOV-4-751	**a.** Perform the following: **1)** Locally unlock and close the breakers: • 40615 for MOV-4-750 • 40731 for MOV-4-751 **2)** Manually close valve(s). IF valve(s) can **NOT** be manually closed, THEN locally close valve(s).
b. Verify MOV-4-869, Safety Injection To Hot Leg Isolation – CLOSED	**b.** Manually close valve. IF valve can **NOT** be manually closed, THEN locally close valve.
c. Verify other paths to outside Containment – CLOSED:	
1) MOV-4-872, Alternate Low Head SI – CLOSED	**1)** Manually close valve. IF valve can **NOT** be manually closed, THEN locally close valve.
2) Locally in Pipe And Valve Room, verify 4-990, MOV-4-869 Bypass Valve – CLOSED	**2)** Locally on the Auxiliary Building roof, close and lock 4-990.
3) Locally in Containment Spray Pump Room, verify 4-895V, SI Test Line Isolation – CLOSED	**3)** Locally close valve.

BASIS DOCUMENT

WOG Procedure Step 1 **PTN Procedure Step** 1

Verify Proper Valve Alignment

BASIS:

This step instructs the operator to verify that all normally closed valves in low pressure lines that penetrate containment are closed. The valving connecting the RHR System to the RCS is of particular interest in this step since the RHR System is a low pressure system (600 psig) connected to the high pressure Reactor Coolant System (2500 psig). Therefore, a rupture or break outside containment is most probable to occur in the low pressure RHR System piping.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

2 At Turkey Point, power is normally removed from RHR loop suction valves. In this condition, Control Room position indication lights are also de-energized. Substep a was modified to use DCS valve position or the indicator lights on the Alternate Shutdown Panel to determine valve positions.

9 Due to the modification of Substep a, a single RNO action is not feasible. Each subsequent substep was provided with an appropriate RNO.

8 The words "low-head SI" were changed to "RHR" to conform with plant specific terminology.

9 Valve numbers were added to aid the operator.

9 MOV-*-869 Bypass Valve, *-990 added per EPU.

2 At Turkey Point the hot leg injection line is on the high-head SI pump discharge. Substep b was reworded to reflect the plant specific system design.

PLANT SPECIFIC SETPOINTS:

N/A

Question # 17

Given the following sequence of events:

- 08:00 a.m.
    - Unit 4 experiences a Safety Injection.
- 10:00 a.m.
    - The crew transitions from 4-EOP-E-1, Loss of Reactor or Secondary Coolant, to 4-EOP-ES-1.3, Transfer to Cold Leg Recirculation.
    - 4A HHSI Pump has tripped and is NOT available.
    - 4B is the ONLY running HHSI Pump.

Which one of the following completes the statements below?

IAW 4-EOP-ES-1.3, prior to realigning the RHR pump suction path, the Operators will start ONE Unit 3 HHSI Pump in order to ______.

A. minimize running Unit 4 HHSI Pumps with only miniflow protection

B. provide sufficient core cooling

C. anticipate potential tripping of the remaining 4B HHSI pump

D. allow HHSI pump 4B suction to be realigned

Answer Analysis

Discussion: Applicant needs to recognize: (1) the basis / reason for establishing TWO HHSI pumps prior to realigning RHR. It is to ensure sufficient core cooling for the transition. Minimizing RWST usage is also a priority but the stem indicates that ONLY ONE Unit 4 HHSI pump is running and TWO are required.

Stopping Unit 3 HHSI pumps is required if Unit 4 RCS pressure is below Unit 4 HHSI pump shutoff head and this action does occur in E-1 prior to step 18 transition to ES 1.3 for SBLOCA responses. The time snapshots are needed to establish that less than 14 hours have occurred since the initiation of the LBLOCA

A Plausible because running Unit 4 HHSI pumps when realigned to the Unit 4 RWST will momentarily place 4B HHSI on only its miniiflow flowpath.

B Correct

C Plausible because the action to realign unit 4 RHR could be seen as potential to trip the 4B HHSI pump due to cavitation / pressure changes in the Unit 4 RWST

D Plausible because ES-1.3 will realign 4B HHSI suction away from the RWST.

Tier: 1 **Group:** 1

K/A: W E11K1.03: Loss of Emergency Coolant Recirculation / 4
Knowledge of the reasons for the responses of: Components, capacity, and function of emergency systems during a Loss of Emergency Coolant Recirculation;

Importance Rating: **3.7**

10 CFR Part 55: 41.5

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must evaluate the conditions and annunciator provided to determine ES 1.3 , Transition to Cold Leg Recirculation is required. Then the applicant must demonstrate knowledge that TWO HHSI pumps are required (reason /basis) to provide sufficient core cooling prior to realigning RHR pumps to the Recirculation sumps as part of the overall mitigating strategy.

SRO Justification: N/A

Technical References: ES 1.3 Transfer to Cold Leg Recirculation
E-1, Loss of Reactor or Secondary Coolant
UFSAR section Chapter 14, Section 14.3 Rupture of a Reactor Coolant Pipe

Proposed references to be provided: None

Learning Objective: PTN 6902330 OBJ. 4

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

BASIS DOCUMENT

W97/fm/ln/cls

WOG Procedure Step N/A **PTN Procedure Step** 6

Check High-Head SI Pump Status

ES 1.3

BASIS:

The status of high-head SI pumps at this point is a function of the type of event in progress.

For large break LOCAs, two high-head SI pumps will be injecting and should both remain in

operation while RHR is secured in subsequent steps.

For smaller break LOCAs, one or more high-head SI pumps may have been secured during SI

reduction. If RCS pressure is below the high-head SI pump shutoff head, at least one high-head

SI pump should be operated to ensure RCS makeup. To provide sufficient core cooling flow,

two HHSI pumps are required up to 14 hours into the LOCA event. After 14 hours decay heat

and core boiling is reduced to the extent that only one HHSI pump is required to be in service.

With pressure greater than the high-head SI pump shutoff head, the pumps would not provide

injection and should not be operated. RHR pumps will be stopped in the next step. This step

ensures availability of SI pumps prior to stopping RHR pumps. Since SI pumps are required

prior to stopping RHR pumps, a check of RCS pressure is required to ensure long term availability of the SI pumps since SI pump recircs will be isolated in subsequent steps.

STEP DEVIATIONS FROM WOG GUIDELINES:

TYPE DESCRIPTION

2 This step provides plant specific actions for establishing cold leg recirculation.

9 Step written with additional clarifying guidance and written as a continuous action step that

if the HHSI pumps are secured they are restarted if pressure drops below the pressure

setpoint.

PLANT SPECIFIC SETPOINTS:

1625 psig Shutoff head pressure of the high-head SI pumps plus normal channel accuracy.

(EOP Setpoint B.5)

1950 psig Shutoff head pressure of the high-head SI pumps plus normal channel accuracy

and post-accident transmitter errors, not to exceed 2000 psig. (EOP Setpoint B.6)

14 hours Time period for reduction from two HHSI pumps to one HHSI pump.

(See CN-LIS-08-142 for more information)

WOG Procedure Step 3 - CAUTION 2 **PTN Procedure Step** 6 - CAUTION 1

If RCS pressure is greater than 1625 psig [1950 psig], High-Head SI Pumps are required to

be stopped.

BASIS:

Because high-head SI pump miniflow lines to the RWST will be isolated when the SI System is

aligned for cold leg recirculation, the pumps will have no miniflow protection. If RCS pressure

increases above their shutoff head, the high-head SI pumps would be dead-headed and could

potentially be damaged.

BASIS DOCUMENT

WOG Procedure Step N/A **PTN Procedure Step** 6

Check High-Head SI Pump Status

BASIS:

The status of high-head SI pumps at this point is a function of the type of event in progress. For large break LOCAs, two high-head SI pumps will be injecting and should both remain in operation while RHR is secured in subsequent steps.

For smaller break LOCAs, one or more high-head SI pumps may have been secured during SI reduction. If RCS pressure is below the high-head SI pump shutoff head, at least one high-head SI pump should be operated to ensure RCS makeup. To provide sufficient core cooling flow, two HHSI pumps are required up to 14 hours into the LOCA event. After 14 hours decay heat and core boiling is reduced to the extent that only one HHSI pump is required to be in service.

With pressure greater than the high-head SI pump shutoff head, the pumps would not provide injection and should not be operated. RHR pumps will be stopped in the next step. This step ensures availability of SI pumps prior to stopping RHR pumps. Since SI pumps are required prior to stopping RHR pumps, a check of RCS pressure is required to ensure long term availability of the SI pumps since SI pump recircs will be isolated in subsequent steps.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

2 This step provides plant specific actions for establishing cold leg recirculation.

9 Step written with additional clarifying guidance and written as a continuous action step that if the HHSI pumps are secured they are restarted if pressure drops below the pressure setpoint.

PLANT SPECIFIC SETPOINTS:

1625 psig Shutoff head pressure of the high-head SI pumps plus normal channel accuracy. (EOP Setpoint B.5)

1950 psig Shutoff head pressure of the high-head SI pumps plus normal channel accuracy and post-accident transmitter errors, not to exceed 2000 psig. (EOP Setpoint B.6)

14 hours Time period for reduction from two HHSI pumps to one HHSI pump. (See CN-LIS-08-142 for more information)

Question # 18

Given the following:

- Unit 3 is experiencing a Loss of Secondary Heatsink.
- AFW flow is NOT available.

Which one of the following completes the statements below?

IAW 3-EOP-FR-H.1, Loss of Secondary Heat Sink, the crew will

secure __(1)__ RCP(s) in order to__(2)__.

A. (1) 1
(2) limit the heat input from RCPs

B. (1) 1
(2) reserve it(them) for future use

C. (1) ALL
(2) limit the heat input from RCPs

D. (1) ALL
(2) reserve it(them) for future use

Answer Analysis

A. Incorrect.
Part 1 incorrect, plausible since 3-EOP-FR-C.2 strategy stops one RCP for later use.
Part 2 correct.

B. Incorrect. plausible since 3-EOP-FR-C.2 strategy stops one RCP for later use.

C. Correct. 3-EOP-FR-H.1. directs to secure all RCPs in order to limit heat input to the RCS, so that the effectiveness of the SG inventory can be extended.

D. Incorrect.
Part 1 correct. Part 2 incorrect, plausible since 3-EOP-FR-C.2 strategy stops one RCP for later use.

Question Number: 18

Tier: 1 **Group:** 1

K/A: (BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink

E05K3.4; Knowledge of the reasons for the following responses as they apply to the (Loss of Secondary Heat Sink)

RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the

facilities license and amendments are not violated.

Importance Rating: **3.7**

10 CFR Part 55: 41.5, 10

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-EOP-FR-H.1, BD-3-EOP-FR-H.1.

Proposed references to be provided: None

Learning Objective: PTN 6902337 OBJ. 4

Cognitive Level: Higher _

Lower X_

Question Source: New

Modified Bank _X

Bank _

Question History: Diablo Canyon 2016

Comments: Screenshot of development references as applicable

BASIS DOCUMENT

PWROG Guideline Step 4 **PTN Procedure Step** 4

Stop All RCPs

BASIS:

RCP operation results in heat addition to the RCS water. By stopping the RCPs, the effectiveness of the remaining water inventory in the S/Gs is extended, which extends the time at which the operator action to initiate bleed and feed must occur. This extension of time is additional time for the operator to restore feedwater flow to the S/Gs.

STEP DEVIATIONS FROM PWROG GUIDELINES:
TYPE DESCRIPTION

N/A

PLANT SPECIFIC SETPOINTS:

N/A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Foldout page is required to be monitored throughout this procedure.

➔ **2. Check If Bleed And Feed Is Required**

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a.	Check the following:	
	1) Two S/G Wide Range Levels – LESS THAN 10% [Narrow Range Level in all S/Gs – LESS THAN 27%]	**1)** Observe CAUTION prior to Step 3, and go to Step 3.
	2) Feedwater Sources – **NONE** ESTABLISHED	**2)** IF CETs can be maintained stable OR decreasing, THEN observe CAUTION prior to Step 3, and go to Step 3.
b.	Stop all RCPs	
c.	Observe CAUTION prior to Step 13, and go to Step 13	

Original for Question # 18.

Exam Bank Question

Facility: Turkey Point

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

The crew is performing the actions of EOP FR-C.2, Response to Degraded Core Cooling. The operator is directed to trip___________RCP(s) in order to____________.

A. 2; limit the heat input from two RCPs during a degraded core cooling condition

B. 1; limit the heat input from one RCP during a degraded core cooling condition

C. 2; preserve two pumps from damage that may occur during operation under highly voided conditions

D. 1; preserve one pump from damage that may occur during operation under highly voided conditions

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Only 1 RCP is stopped. Limiting heat input is an issue for FR-H.1.

B. Incorrect. Limiting heat input is an issue for FR-H.1.

C. Incorrect, Only 1 RCP is stopped.

D. Correct. 1; preserve one pump from damage that may occur during operation

Original for Question # 18.

under highly voided conditions

Techical Reference(s): LPE-C, FR-C.2 background, H.1 background, E-0.2 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: 7920M - Explain basis of emergency procedure steps (FR-Cs). (As available)

Question Source: Bank 21714

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2016 Diablo Canyon

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Question # 19

Given the following:

- Unit 3 is at 72% reactor power.
- A ramp up to 100% and associated dilution are in progress.
- Tavg is established at 3°F higher than Tref.

Subsequently:

- ANN B7/1, NIS/RPI ROD DROP ROD STOP, alarms.
- Tavg is now 1°F higher than Tref.

Which one of the following completes the statement below?

IAW 3-ONOP-028.3, Dropped RCC, the crew is required to NEXT ______.

A. MANUALLY trip the reactor.

B. LOWER power to < 50% reactor power

C. REDUCE the Power Range NIs Hi Flux Trip Setpoint

D. LOWER Turbine Load to return Tavg to program

Answer Analysis

Discussion: Answer B. The RO applicant must recognize the conditions of the single dropped rod, the RCS temperature and power at the time and determine that Axial Flux is ok. Based on that rods need to be placed in manual and power reduced to< 50 % within 60 minutes.

A Plausible, since multiple dropped rods would require a reactor trip IAW 3-ONOP-028.3.

B Correct.

C Plausible because this is an action directed by 3-ONOP-028.3 later on.

D Plausible because ONOP 028.3 would require reducing Turbine load if Tavg were low. Reducing load would exacerbate the overtemperature condition

Question Number: 19

Tier: 1 Group 2

K/A: 003G2.1.7: Ability to evaluate plant performance and make operational judgements related to "Dropped control rod" based on operating characteristics, reactor behavior and instrument interpretation

Importance Rating: **4.4; 10 CFR Part 55:** 41.5

10 CFR 55.43.b :N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, identify the dropped rod condition and determine operator action required to mitigate the event per the dropped rod 0NOP

SRO Justification: N/A

Technical References: 3-ONOP 028.3 , Dropped RCC

Proposed references None

Learning Objective: PTN 6902207 OBJ. 6

Cognitive Level: Higher X_

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

For reducing power to less than 50% in the following steps, the NI may not be accurate due to the dropped rod. Power should be reduced until the Core Δ T Power, as indicated on DCS or per the plant curve book, is less than 50%.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Check AFD Within RAOC** • G 5/1, AXIAL FLUX T.S. LIMIT EXCEEDED - OFF • At least 3 channels of AFD indicating within the RAOC limit as defined in the Plant Curve Book, Section 5, Figure 1	Within 30 minutes, reduce reactor power to less than 50% as indicted on core ΔT using 3-GOP-100, FAST LOAD REDUCTION, while continuing with this procedure.
6	**Initiate Hourly QPTR Determination Using 3-OSP-059.10, DETERMINATION OF QUADRANT POWER TILT RATIO Until Either QPTR Results Are Satisfactory <u>OR</u> Reactor Power Is Less Than 50%, as indicated on core ΔT**	
7	**Declare The Dropped RCC Inoperable**	
8	**Reduce Reactor Power To Less Than 50% Within 1 Hour** a. Within one hour, reduce reactor power to less than 50% as indicted on core ΔT using 3-GOP-100, FAST LOAD REDUCTION, while continuing with this procedure	

Question # 20

Given the following:

- Unit 3 is at 70% power.
- One Control Bank D RCC is determined to be stuck (untrippable).
- All RCCS are within the Allowed Rod Misalignment.

Which one of the following completes the statements below?

IAW Tech Specs __(1)__ is required to be calculated within a MAXIMUM

of __(2)__ minutes.

A. (1) SHUTDOWN MARGIN
(2) 15

B. (1) SHUTDOWN MARGIN
(2) 60

C. (1) QPTR
(2) 15

D. (1) QPTR
(2) 60

Answer Analysis

A. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since other Tech Spec actions have a 15 minute time limit, i.e. TS 3.2.1 restore AFD to RAOC limits.

B. Correct. IAW TS 3.1.3.1 Action A.

C. Incorrect.
Part 1 incorrect, plausible since this is required for other control rod failures, i.e. dropped rod.
Part 2 incorrect, plausible since other Tech Spec actions have a 15 minute time limit, i.e. TS 3.2.1 restore AFD to RAOC limits.

D. Incorrect.
Part 1 incorrect, plausible since this is required for other control rod failures, i.e. dropped rod.
Part 2 correct.

Question Number: 20

Tier: 1 **Group:** 2

K/A: 000005 (APE 5) Inoperable/Stuck Control Rod

005AK1.05; Knowledge of the operational implications of the following concepts as

they apply to Inoperable / Stuck Control Rod:

Calculation of minimum shutdown margin

Importance Rating: **3.3**

10 CFR Part 55: 41.8, 10

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must determine the effect of a stuck control rod on shutdown margin

SRO Justification: N/A

Technical References: TS 3.1.3.1

Proposed references None

to be provided:

Learning Objective: PTN 6902207 OBJ. 4

Cognitive Level: Higher _

Lower X_

Question Source: New

Modified Bank X

Bank _

Question History: Turkey Point 2011

Comments:

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within the Allowed Rod Misalignment between the Analog Rod Position Indication and the group step counter demand position within one hour after rod motion. The Allowed Rod Misalignment shall be defined as:

a. for THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 18 steps, and

b. for THERMAL POWER greater than 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 12 steps.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps and THERMAL POWER greater than 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER and confirm that all indicated rod positions are within the Allowed Rod Misalignment, or

   3. Be in HOT STANDBY within the following 6 hours.

c. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 18 steps and THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Be in HOT STANDBY within the following 6 hours.

---

* See Special Test Exceptions 3.10.2 and 3.10.3.

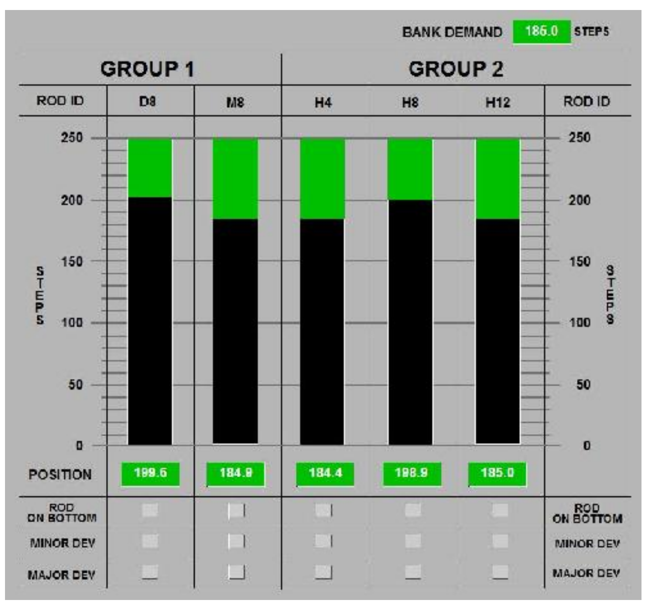
Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	005		AA2.03
	Importance Rating	3.5		

Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable

Proposed Question: RO Question # 27

Given the following conditions:

- A down power is in progress on Unit 3.
- Reactor power is 75%.
- Control rods D8 and H8 are mechanically bound as follows:



ALSO PROVIDED AS A REFERENCE IN LARGER FORMAT

In accordance with TS 3.1.3.1, Movable Control Assemblies, which one of the following identifies the MINIMUM required action within ONE hour?

A.	Calculate shutdown margin IAW 0-OSP-028.8, Shutdown Margin Calculation.
B.	Log rod heights IAW 3-OSP-201.1, RO Daily Logs.

C.	Calculate QPTR IAW 3-OSP-059.10, Quadrant Power Tilt Ratio Calculation.
D.	Verify AFD IAW 3-OSP-059.9, Computer Axial Flux Monitor Verification.

Proposed Answer: A

A.	CORRECT. Technical Specifications 3.1.3.1, *Movable Control Assemblies – Group Height*, Action a, requires SDM verification within 1 hour.
B.	Incorrect: plausible since requirement for OOS ANN B9/3
C.	Incorrect since a QPTR verification is not required for an inoperable rod. Plausible since rod misalignment can cause power peaks within the core and severe misalignment can cause QPTR to exceed the limits of TS 3.2.4.
D.	Incorrect since an AFD verification is not required for an inoperable rod. Plausible since, per 0-ADM-536, *Technical Specification Bases Control Program*, Page 37, rod alignment is related to AFD. Also plausible since some AFD T.S. Actions are less than 1 hour.

Technical Reference(s)	Tech Spec 3.1.3.1, Movable Control Assemblies - Group Height	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:	NO

Learning Objective:	LP 6902521, Obj. 3	(As available)

Question Source:	Bank	12886	
	Modified Bank	X	(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	2011	Turkey Point

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	5
	55.43	

Question # 21

Which one of the following completes the statements below?

IAW 4-EOP-FR-S.1, Response To Nuclear Power Generation/ATWS, the preferred boration flowpath is through __(1)__.

IAW BD-EOP-FR-S.1, Basis Document for 4-EOP-FR-S.1, if reactor power is greater than 5% the RCPs are NOT to be tripped in order to __(2)__.

A. (1) MOV-4-350, Emergency Boration Valve
   (2) maintain core heat removal

B. (1) MOV-4-350, Emergency Boration Valve
   (2) provide mixing for the boration

C. (1) FCV-4-113B, Boric Acid to Blender
   (2) maintain core heat removal

D. (1) FCV-4-113B, Boric Acid to Blender
   (2) provide mixing for the boration

Answer Analysis

Discussion: Answer A. Discussion: Emergency Boration is for an ATWS condition. The preferred sequence is : MOV – 350, then Normal boration with Max flow. IAW 3-FR-S.1, RCPs are required to be maintained in operation to ensure proper core cooling is maintained.

A Correct

B. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since, running RCPs will mix the RCS and the boration.

C. Incorrect.
Part 1 incorrect, plausible since this is a possible boration flowpath.
Part 2 correct.

D. Incorrect.
Part 1 incorrect, plausible since this is a possible boration flowpath.
Part 2 incorrect, plausible since, running RCPs will mix the RCS and the boration.

Question Number: 21

Tier: 1 Group 2

K/A: 024AK3.02: **Knowledge of the reasons for** Actions contained in EOP for emergency boration **as they apply to Emergency Boration:**

Importance Rating: **4.2; 10 CFR Part 55:** 41.5

10 CFR 55.43.b :N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, identify the

SRO Justification: N/A

Technical References: BD FRS.1, ATWS

Proposed references None

Learning Objective:

Cognitive Level: Higher _

Lower X_

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

BASIS DOCUMENT

WOG Procedure Step 1 - CAUTION **PTN Procedure Step** 1 - CAUTION

RCPs should NOT be tripped with reactor power greater than 5%.

BASIS:

During an ATWS, RCP operation could be beneficial by temporarily cooling the core under voided RCS conditions. If reactor power is greater than 5%, the RCPs should not be tripped even if all normal running conditions are not satisfied. Manually tripping the RCPs during some ATWS events could result in reduced heat removal and a challenge to fuel integrity. An ATWS is not a design basis event; therefore, the licensing requirement to trip the RCPs within a timely manner to remain within the small-break LOCA design basis is not applicable.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

N/A

PLANT SPECIFIC SETPOINTS:

5% Maximum nuclear power level for performance of EOP network. (EOP Setpoint P.2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**(continued)**	
e.	Open MOV-4-350, Emergency Boration Valve	**e.** Perform the following: **1)** Open FCV-4-113A, Boric Acid To Blender. **2)** Open FCV-4-113B, Blender Flow To Charging Pump. **3)** Locally open 4-356, Manual Emergency Boration Valve. **4)** WHEN 4-356, Manual Emergency Boration Valve is open, THEN close FCV-4-113B, Blender To Charging Pump. **5)** Continue with Step 4.f.
f.	Open HCV-4-121, Charging Flow To Regen Heat Exchanger	
g.	Verify CV-4-310A, Loop A Charging Isolation – OPEN	**g.** Open CV-4-310B, Loop C Charging Isolation.
h.	Establish Emergency Boration flow: • FI-4-110 – GREATER THAN 60 GPM, OR 0 GPM IF MOV-4-350 IS CLOSED • FI-4-122A – GREATER THAN 45 GPM	**h.** Perform one or more of the following as necessary to establish Emergency Boration flow: * Adjust operating Charging Pump(s) speed controller(s). * Start additional Charging Pumps. * Manually align valves.

Question # 22

Given the following:

- Unit 3 is at 100% power.
- Channel Select Pressurizer Level Control is in the CH. 1 & 2 position.

Subsequently:

- ANN A8/4, PZR LO-LO LEVEL ALERT, alarms.
- LI-3-461, PZR Level Prot/Control, indicates 0%.

Which one of the following completes the statements below?

FI-3-122, Charging Line Flow, indication will __(1)__.

LCV-3-460, High Pressure LTDN Isol. Vlv,__(2)__ AUTOMATICALLY close.

A. (1) remain the same
(2) will

B. (1) remain the same
(2) will NOT

C. (1) raise
(2) will

D. (1) raise
(2) will NOT

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

Since the Control Channel Select Switch is in CH. 1 & 2 position and LT-461 has failed low as indicated by the LI-461 indication and the annunciator A-8/4, this will not cause any automatic action since it is not in the control circuit (LT-456 is Channel I/Path A, and LT 460 is Channel II/Path B)

A. (1) Correct, Since the Control Channel Select Switch is in CH. 1 & 2 position, LT-461 will not cause any automatic action since it is not in the control circuit (LT-456 is Channel I/Path A, and LT 460 is Channel II/Path B)
(2) Incorrect; Plausible if the candidate does not realize that based on the CH Select Switch position that there will not be any automatic action for this failure.

B. (1) Correct,
(2) Correct,

C. (1) Incorrect, Plausible if the candidate does not realize that LT-461 is not aligned to the portion of the circuit that controls charging pump flow.
(2) Correct

D. (1) Incorrect, see C(1)
(2) Incorrect, see A(2)

Question Number: 22

Tier: 1 **Group:** 2

K/A: 000028 (APE 28) Pressurizer (PZR) Level Control Malfunction

028AK2.02; Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following:

Sensors and detectors

Importance Rating: **2.6**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 5610-T-D-15

Proposed references to be provided: None

Learning Objective: PTN 6902113 OBJ. 6

Cognitive Level: Higher X _

Lower _

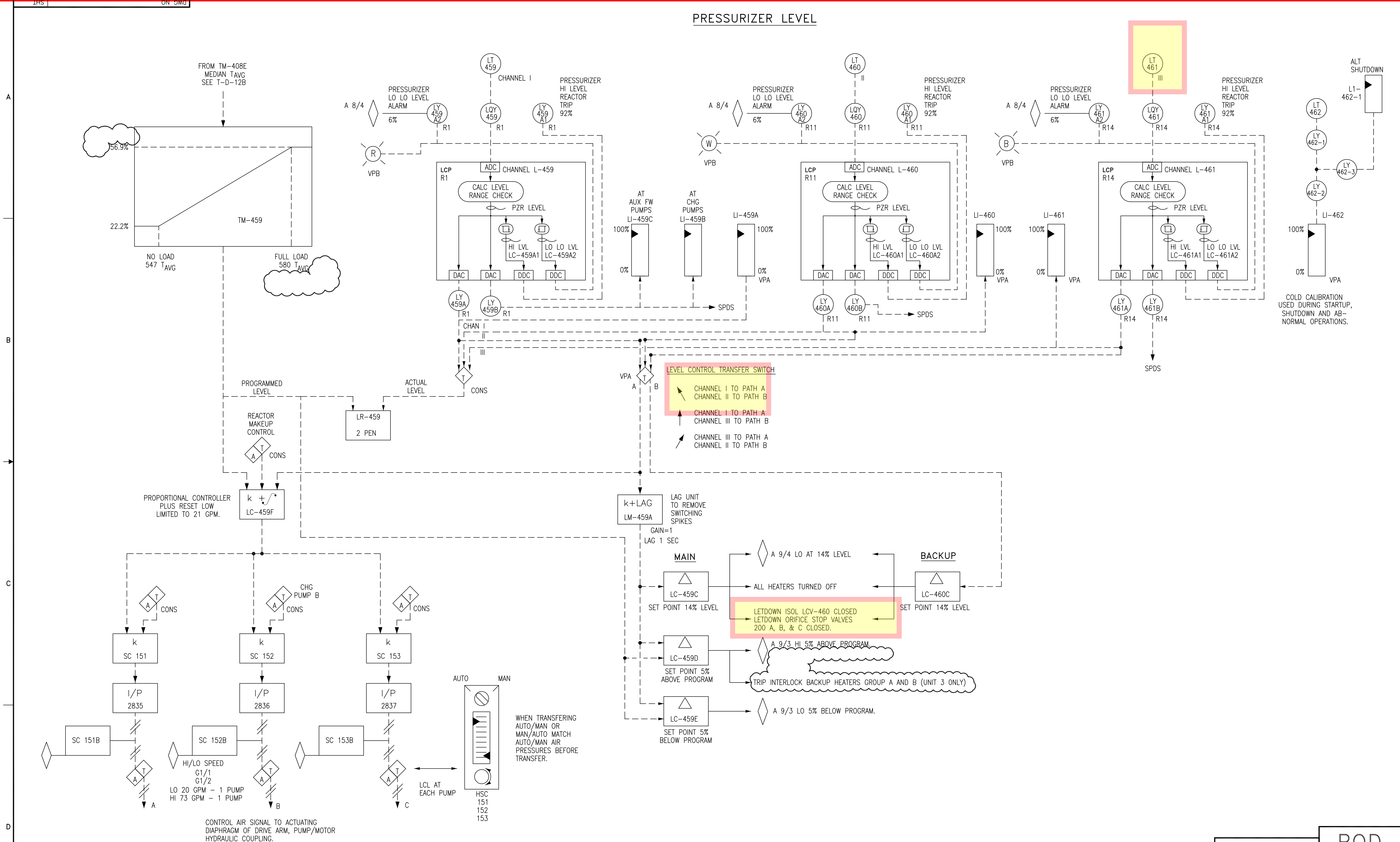
Question Source: New _

Modified Bank X_

Bank

Question History: Original question used on PTN 2017-301 exam. RO question number 20. Original KA 028 AA2.03. Question modified to change the mode of failure From PSR control input relay to PZR Level Control instrument failure.

Comments: Screenshot of development references as applicable



REV	DATE	REVISION	BY	CH	APP	APP
24	07-10-13	ISSUED AS-BUILT PER EC'S 247049, 246978, 246977; INCORP. CRN'S 002, 017 AND 023 (ITOP'S 13-07-008, 015 & 017).	RV	BB	–	PMB
23	1-5-13	ISSUED AS-BUILT PER CRN-017 (EC 246977) ITOP 13-01-013.	AFG	BB	–	PMB
22	7-29-12	ISSUED AS-BUILT PER EC 247048 (PC/M 10-039).	RH	BB	TK	JD

REV	DATE	REVISION	BY	CH	APP	APP
21	6/26/09	ISSUED AS-BUILT PER CRN I-5048 (PC/M 09-034).	RV	RH	–	TS
20	10-21-97	ISSUED AS-BUILT PER CRN-I-3593(PC/M 97-028).	RV	RH	BAP	TS
19	5/11/95	ISSUED AS-BUILT FOR PC/M 94-120	RV	RH	BAW	JCW
18	3/2/93	AS-BUILT PER DCR TPE-92-316	RH	RV	ET	JCW
17	5/8/92	AS-BUILT PER DCR TPI-91-463	RH	RV	ET	JCW

QUALITY RELATED POD

TURKEY POINT NUCLEAR UNITS 3 & 4

ELECTRICAL

FLORIDA POWER & LIGHT

PRESSURIZER LEVEL CONTROL AND PROTECTION CHARGING PUMP CONTROL

DRAWING NUMBER 5610-T-D-15

SYS

REV 24

SHEET 1 OF 1

DWG NO 5610-T-D-15 SHT 1

5610-T-D-15_SH 1.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	028		AA2.03
	Importance Rating	2.8		

Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Charging subsystem flow indicator and controller

Proposed Question: RO Question # 20

Giving the following conditions:

- Unit 3 is at 100% power.

Subsequently:

- LC-459CX, PRZ Level Comparator Relay de-energizes.

Which one of the following describes the expected plant response?

FI-3-122, Charging Line Flow, indication will __(1)__.

The controller demand for HCV-3-121, Charging flow to Regen Heat Exchanger, __(2)__ AUTOMATICALLY change.

A.	(1) remain the same (2) will
B.	(1) remain the same (2) will NOT
C.	(1) lower (2) will
D.	(1) lower (2) will NOT

Proposed Answer: D

A.	**Incorrect** Both parts are incorrect, but plausible per discussion below. Question parts are independent and plausible as a whole.
B.	**Incorrect** Part 1 is incorrect, but plausible since the PRZ level control system is intact.
C.	**Incorrect** Part 2 is incorrect, but plausible since charging flow lowers and other systems control flow by positioning control valves, ie. main FRV.
D.	**Correct** LC-549CX causes LD to isolate, PRZ Level rises and Charging speed is reduced, this causes charging flow to lower. HCV-3-121 can only be adjusted manually.

Technical Reference(s)	3-ONOP-041.6		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	LP 6902254 Obj 6		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question # 23

Given the following:

- Unit 3 is at 5% reactor power .

Subsequently

- N-3-35, Intermediate Range NI, loses compensating voltage.
  The Shift Manager declares N-3-35 INOPERABLE.

Which one of the following completes the statements below?

The N-3-35, Level Trip Switch is located on the __(1)__.
IAW 3-ONOP-059.7, Intermediate Range Nuclear Instrument Malfunction, the crew __(2)__ required to take the N-3-35, Level Trip Switch to Bypass.

A. (1) Control Room Console
   (2) is

B. (1) Control Room Console
   (2) is NOT

C. (1) N-3-35 Drawer
   (2) is

D. (1) N-3-35 Drawer
   (2) is NOT

Answer Analysis

Discussion: Answer C. Per 3-ONOP-059.7 each IR Nuclear instrument failure will require placing the Level Trip switch in Bypass.

A Incorrect, see B

B 1) Plausible, since other manipulations associated with this failure take place on the console, switching the NI recorder.
2) Plausible, other NI failures do NOT require this type of manipulation, i.e. loss of Power Range NI.

C 1) Correct
2) Correct

D Incorrect, See B

Question Number: 23

Tier: 1 Group 2

K/A: 033AA1.02: **Ability to operate or monitor the** Level Trip Bypass **as it applies to a loss of Intermediate Range Nuclear Instrumentation:**

Importance Rating: **3.0; 10 CFR Part 55:** 41.7

10 CFR 55.43.b **:**N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, identify the need to place both IR channels in Level Trip bypass , one due to loss of compensating voltage and one due to loss of power.

SRO Justification: N/A

Technical References: ARP B 8/4 NIS Trip Bypassed
3-0nop 059.7 Intermediate Range Nuclear Instrument Malfunction, steps 4-6

Proposed references None

Learning Objective: PTN 6902206 OBJ. 4

Cognitive Level: Higher _

Lower X_

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Placing the IR channel level trip switch in BYPASS will cause annunciator B 8/4 NIS TRIP BYPASSED to alarm.

4 **Align Failed Channel**

a. Place level trip switch on affected IR channel in BYPASS

b. Check one IR Channel Operable

RESPONSE NOT OBTAINED:

b. Perform the following:

1) **WHEN** below P-10, **THEN** restore at least one channel to Operable within 1 hour **OR** place the Unit in Hot Standby within the next 6 hours. (Tech Spec 3.0.3).

a) **WHEN** Backup NIS (Gamma Metrics) Source Range indicates less than 10^4 CPS, **THEN** simultaneously depress both P-6 DEFEAT pushbuttons on console to energize Source Range.

2) Switch console recorder to operable NIS channel.

3) Go to Step 5.

c. Switch console recorder to monitor operable IR channel

5 **Check Reactor In Mode 1 Above P-10**

RESPONSE NOT OBTAINED: Go to Step 6.

a. Ensure IR High Level Rod Stop is blocked using 3-GOP-301, HOT STANDBY TO POWER OPERATION

RESPONSE NOT OBTAINED: a. **IF** IR High Level Rod Stop will not block, **THEN** notify I&C.

b. Go to Step 10

6 **Check Reactor In Mode 1 Below P-10**

RESPONSE NOT OBTAINED: Go to Step 7.

a. Maintain reactor power less than 10 percent until both IR channels are operable (Tech Spec 3.3.1)

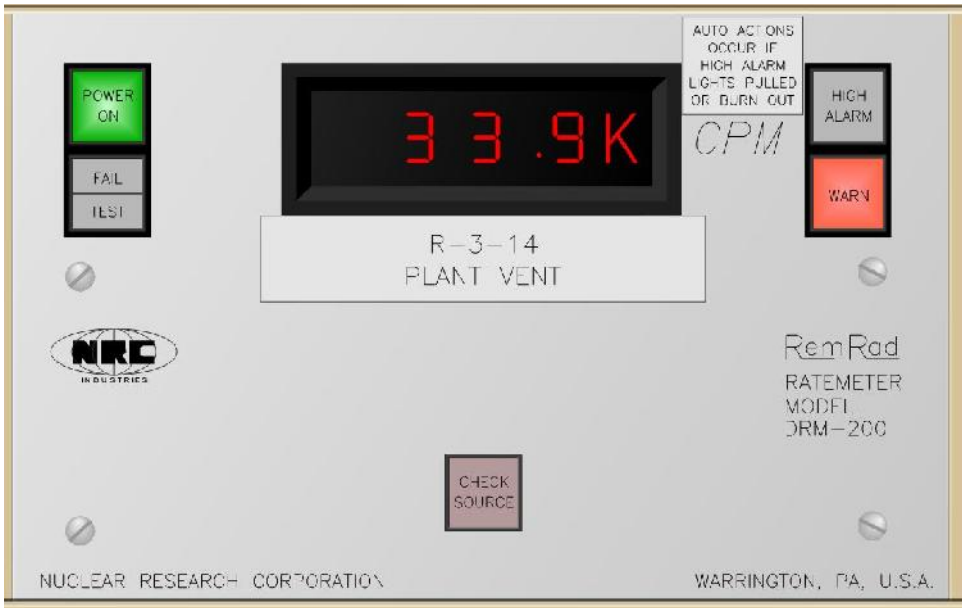
b. Go to Step 10

Question # 24

Given the following:

- Both Units are at 100% power.
- A release of Gas Decay Tank A is in progress.

Subsequently R-14, Plant Vent PRMS indicates as follows:



ALSO PROVIDED AS A REFERENCE IN LARGER FORMAT

Which one of the following completes the statements below?

3-ONOP-067, Radioactive Effluent Release, entry criteria __(1)__ met.

RCV-014 __(2)__ expected to have AUTOMATICALLY CLOSED.

A. (1) is
(2) is

B. (1) is
(2) is NOT

C. (1) is NOT
(2) is

D. (1) is NOT

(2) is NOT

Answer Analysis :

Discussion of system operation and conditions that relates to the correct answer choice.

A. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since this would be the correct response for a HIGH ALARM .

B. Correct
(1) Correct, IAW 3-ONOP-067 an alarm on R-3-14 meets the entry criteria into this procedure.
(2) Correct, since there is only an Alert Warning and not an Alarm on R-14, the valve will have to be closed by procedure.

C. Incorrect.
Part 1 incorrect, plausible since the high alarm, which causes the automatic actions, is not received.
Part 2 incorrect, plausible since this would be the correct response for a HIGH ALARM .

D. Incorrect.
Part 1 incorrect, plausible since the high alarm, which causes the automatic actions, is not received.
Part 2 incorrect, plausible since this would be the correct response for a HIGH ALARM .

Question Number: 24

Tier: 1 **Group:** 2

K/A: 000060 (APE 60) Accidental Gaseous Radwaste Release

060AA2.05. Ability to determine and interpret the following as they apply to

the Accidental Gaseous Radwaste:

That the automatic safety actions have occurred as a result of a high ARM system signal

Importance Rating: **3.7**

10 CFR Part 55: N/A

10 CFR 55.43.b **:** 43.5

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-ONOP-067

Proposed references to be provided: None

Learning Objective: PTN 6902242 OBJ. 2

Cognitive Level: Higher X _

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

1.0 PURPOSE

1.1 This procedure provides instructions to be followed in the event of a radiation monitor alarm or malfunction of R-11, R-12, R-14, R-17A/B, R-18, and R-20 that could result in a radiation hazard or inadvertent release of radioactivity to the environment.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 Alarms on following PRMS channels

2.1.1 Containment Air Monitors R-3-11 and R-3-12

2.1.2 Plant Vent Gas Monitor, R-3-14

2.1.3 Component Cooling Water Monitors, R-3-17A/17B

2.1.4 Waste Disposal System Liquid Effluent Monitor R-3-18

2.1.5 Reactor Coolant Letdown Line Radioactivity Monitor, R-3-20

2.2 High radiation alarms on following radiation monitors

2.2.1 Plant Vent SPING, RAD-6304

2.2.2 Unit 3 Spent Fuel Pit Vent SPING, RAD-6418

2.2.3 Area Radiation Monitor

2.3 Control Room Annunciators

2.3.1 H 1/4, PRMS HI RADIATION

2.3.2 H 1/6, PRMS CHANNEL FAILURE

2.3.3 H 3/6, PRMS R11/R12 BYPASSED/WARNING ACTUATED

2.3.4 X 4/1, ARMS HI RADIATION

2.4 An unexplained decrease in level/pressure in tanks containing radioactive liquid or gas.

2.5 Any uncontained spillage of contaminated or potentially contaminated liquids.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If more than one high radiation event is occurring, the operator should prioritize actions to minimize offsite dose.

NOTES

- *Prioritization should include consideration of release rate, size of leak, isolable or NOT, etc.*
- *Step 3 RNO actions should be performed in the determined order of priority.*

3 **Check PRMS High Alarm - OFF**

- Check R-11 **AND** R-12 High Alarms - OFF
- Check R-17A **AND** R-17B High Alarms - OFF
- Check R-14 High Alarm - OFF
- Check R-18 High Alarm - OFF
- Check R-20 High Alarm - OFF

RESPONSE NOT OBTAINED:

Perform the following:

* **IF** R-11 **OR** R-12 High Alarm is ON, **THEN** go to Step 16.
* **IF** R-17A **OR** R-17B High Alarm is ON, **THEN** go to Step 29.
* **IF** R-14 High Alarm is ON, **THEN** go to Step 42.
* **IF** R-20 High Alarm is ON, **THEN** perform 3-ONOP-041.4, EXCESSIVE REACTOR COOLANT SYSTEM ACTIVITY, while continuing with this procedure.
* **IF** R-18 High Alarm is ON, **THEN** perform the following:

  a. Verify RCV-018 - Closed.

  b. **IF** a Liquid Release is in progress, **THEN** terminate the release.

  c. Inform the Shift Manager of R-18 alarm.

  d. Determine and correct the cause of the R-18 high alarm before commencing another liquid release.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

42 **Check For Release To Atmosphere**

a. Verify RCV-014 - CLOSED

a. Perform the following:

1) Close RCV-014 by reducing the hand loader pressure to zero.

2) Verify the valve stem position indicator on RCV-014 indicates the valve is fully closed.

3) **IF** RCV-014 is not fully closed, **THEN** ensure the following GDT valves CLOSED:

- 4638A, A GDT to Plant Vent
- 4638B, B GDT to Plant Vent
- 4638C, C GDT to Plant Vent
- 4638D, D GDT to Plant Vent
- 4638E, E GDT to Plant Vent
- 4638F, F GDT to Plant Vent

b. Check if High Alarm on monitor caused by excessive release rate of gas decay tank

b. Go to Step 42f.

c. Check count rate on all stack monitors - DECREASING

c. Go to Step 42f.

d. Perform following prior to recommencing gas release

1) Notify the Shift Manager and Chemistry of problem with gas release

2) Resample affected gas decay tank

3) Resubmit gas release permit to the Shift Manager for approval

e. Return to Step 1

f. Check following:

- All gas decay tank pressures less than 100 psig
- No gas decay tank pressure decreasing in an uncontrolled manner

f. Perform the following:

1) Isolate the affected tank.

2) Transfer contents of affected tank to another gas decay tank using 0-NOP-061.15A, WASTE GAS COMPRESSORS and 0-NOP-061.15B, WASTE GAS DECAY TANKS.

Question # 25

Given the following:

- Unit 4 SFP shuffle is in progress.

Subsequently:

- ANN X4/1, ARMS HIGH RADIATION, alarms.
- RI-4-1407B, Unit 4 Spent Fuel Pit Canal Area, High Alarm is lit.

Which one of the following completes the statements below?

IAW 0-ONOP-066, High Area Radiation Monitoring System Alarm, the RO will prioritize to __(1)__.

Rising indication on, R-14, Plant Vent Stack PRMS, __(2)__ confirm the high radiation condition in the Unit 4 SFP.

A. (1) make a plant page announcement about the SFP radiological conditions
(2) will

B. (1) make a plant page announcement about the SFP radiological conditions
(2) will NOT

C. (1) notify RP to survey the SFP area
(2) will

D. (1) notify RP to survey the SFP area
(2) will NOT

Answer Analysis

A. Correct. 0-ONOP-066, prioritizes to announce the radiological conditions over the RP surveying activities. Unit 4 SFP is aligned to the plat vent stack.

B. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since, not all SFP exhausts are aligned to the Plant stack, i.e. Unit 3 SFP.

C. Incorrect.
Part 1 incorrect, plausible since this is part of the actions of 0-ONOP-066.
Part 2 incorrect, plausible since, not all SFP exhausts are aligned to the Plant stack, i.e. Unit 3 SFP.

D. Incorrect.
Part 1 incorrect, plausible since this is part of the actions of 0-ONOP-066.
Part 2 correct.

Question Number: 25

Tier: 1 Group: 2

K/A: 061G2.4.45: 4.1**, 41.10 Ability to prioritize and interpret the significance of each annunciator or alarm.**

Importance Rating: 4.1

10 CFR Part 55: 41.10

10 CFR 55.43.b n/a – RO question because question is related to the overall mitigation strategies and the purpose of the ONOP -067

K/A Match: The K/A is matched as the applicant must have knowledge of the ONOP-067 related to the ARMS and PRMS radiation monitoring systems. Specifically, that overall mitigating strategy requires prioritization for multiple radiation alarms and that the Letdown line R-20 monitor requires aligning control room ventilation in emergency recirculation mode

Technical **ONOP-067,** Radioactive Effluent Release, including the foldout page

References: **Lesson Plan:**

Proposed references to be provided: None

Learning Objective:

Cognitive Level: Higher X_

Lower _

Question Source: New x_

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Identify Alarming ARMS Channel(s)** Identify individual channel(s) HIGH alarm light illuminated on ARMS control panel R-30	

NOTE

The alarm should not be silenced until personnel have been evacuated from the affected area and entry to the area has been restricted OR the alarm has been determined to be an invalid alarm.

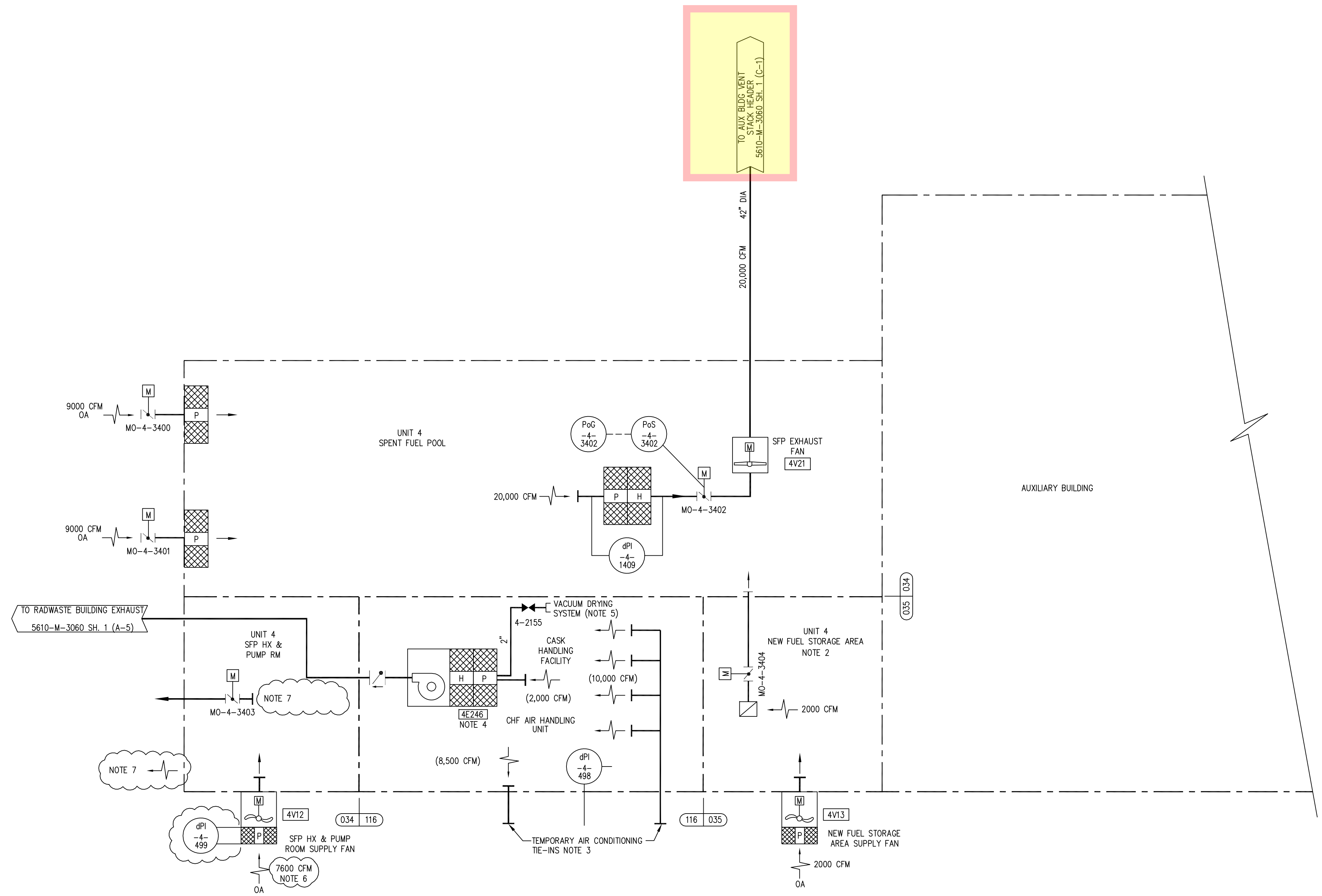
STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	**Verify Alarm Is Valid**	
	a. Press HIGH alarm pushbutton on ARMS control panel to read alarm setpoint	
	b. Digital display matched high alarm setpoint on Attachment 1	b. Perform the following: 1. Press the ALARM ACK pushbutton on ARMS control panel R-30 to silence the alarm. 2. Announce that access has been restored to the evacuated area. 3. Notify RP to install a portable area radiation monitor which meets 10CFR50.68(b) requirements if affected channel is 21 or higher. 4. **IF** any ARM channel 1 through 6 fails while in Mode 5 or 6, **THEN** notify Radiation Protection to install a portable area radiation monitor with alarm 5. Initiate a PWO to repair the defective ARMS channel. 6. Return to procedure and step in effect.

CAUTION

If special fire protection requirements (fire watch etc.) are required for the building or area being evacuated, the fire protection requirements should be re-established as soon as plant conditions allow.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Make Plant Announcement**	
	a. Use the plant page system with page boost	
	b. Announce the affected areas	
	c. Notify people to keep clear	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Check Alarming Channel NOT Inside Containment**	**IF** people are inside containment, **THEN** sound Containment Evacuation Alarm for affected unit.
5	**Notify RP To Survey the Area To Determine The Source Of Radiation**	
	a. RP survey of affected area indicates an increased radiation level	a. Perform the following: 1. Press the ALARM ACK pushbutton on ARMS control panel R-30 to silence the alarm. 2. Announce that access has been restored to the evacuated area. 3. Discontinue area survey. 4. Notify RP to install a portable area radiation monitor which meets 10CFR50.68(b) requirements if affected channel is 21 or higher. 5. **IF** any ARM channel 1 through 6 fails while in Mode 5 or 6, **THEN** notify Radiation Protection to install a portable area radiation monitor with alarm 6. Initiate a PWO to repair the defective ARMS channel. 7. Return to procedure and step in effect.
	b. Coordinate with the RPSS on possible evacuation routes and changing radiological conditions.	
6	**Check ONLY One ARM Channel For The Affected Area Indicates Increased Radiation Levels**	Perform the following: a. Evacuate the area. b. Notify RP of increased radiation levels on ARMS channels. c. Notify Security to restrict entry to the affected area(s).
7	**Press ALARM ACK Pushbutton On ARMS Control Panel**	



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2.
2. NEW FUEL STORAGE COMPONENTS ARE ALL SYSTEM 035.
3. TIE-INS PROVIDED FOR TEMPORARY AIR CONDITIONING DURING ISFSI CAMPAIGNS. TIE-INS ARE NORMALLY CAPPED WHEN CASK HANDLING FACILITY IS NOT IN USE. ASSOCIATED FLOWRATES IN PARENTHESES ONLY APPLY WHEN TEMPORARY AIR CONDITIONING IS CONNECTED.
4. AIR HANDLING UNIT IS ONLY USED DURING ISFSI CAMPAIGNS. UNIT DAMPER IS CLOSED WHEN NOT IN USE.
5. VACUUM DRYING SYSTEM IS ONLY USED DURING ISFSI CAMPAIGNS. UNIT IS HOOKED UP, 4-2155 REQUIRES OPENING AND AHU SHALL BE RUNNING.
6. FLOW RATE AT 70°F AMBIENT TEMPERATURE.
7. 7600 CFM AIR SUPPLIED TO THE HX ROOM EXFILTRATES THROUGH THE TWO GRATED HX ROOM DOORS AND MO-4-3403.

THIS DRAWING SUPERSEDES DRAWINGS:
5610-M-8 REV. 19

REFERENCE DRAWINGS:
5610-M-3060 SH. 1 AUXILIARY BUILDING VENTILATION

NON-NUCLEAR SAFETY

POD

REV	DATE	REVISION	BY	CH	APP	APP
4	8-12-13	ISSUED AS-BUILT PER EC 246921 & INCORP. CRN-160. (ITOP 13-08-022)	RH	RV	CP	JD
3	3-31-12	ISSUED AS-BUILT PER EC 242213 (PC/M 06-019) & INCORP. CRN-012.	RH	RV	BB	SB
2	11-21-96	ISSUED AS-BUILT PER CRN'S I-3462 & I-3464 (PC/M 96-064).	RV	RH	BP	TS
1	4-30-93	INCORP. CRN-M-5813 FOR PC/M 93-002	GM	AB	LIK	JRH
0	7-22-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	TAM	MD	RAS/JGR	LRB
——	——	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-248.	——	——	——	——

FPL

TURKEY POINT NUCLEAR UNIT 4

P & I D

SPENT FUEL POOL
AND NEW FUEL
STORAGE AREA VENTILATION

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5614-M-3034

SYS 034

SHEET 1

REV 4

30x42 E SIZE 04/27/93

Question # 26

Given the following:

- 0-ONOP-105, Control Room Evacuation, is in progress.
- The Unit 3 RO is stationed at the Alternate Shutdown panel.
- Plant cooldown is in progress.
- Unit 3 is at EOL.

Subsequently:

If the ASP __(1)__ indicates source range counts rising, the crew is required to__(2)___ RCS temperature while lining up to commence boration.

A. (1) NI-3-6649, Gammametrics Backup NIS,
(2) lower

B. (1) NI-3-6649, Gammametrics Backup NIS,
(2) raise

C. (1) N-3-32, Source Range NI,
(2) lower

D. (1) N-3-32, Source Range NI,
(2) raise

Answer Analysis

A. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since with a positive MTC a plant cooldown would add negative reactivity.

B. Correct
Gammametrics indication is present at the ASP. Allowing the RCS to heatup will add negative reactivity at EOL due to negative MTC.

C. Incorrect.
Part 1 incorrect, plausible since, the SRNIs also indicate source range counts.
Part 2 incorrect, plausible since with a positive MTC a plant cooldown would add negative reactivity.

D. Incorrect.
Part 1 incorrect, plausible since, the SRNIs also indicate source range counts.
Part 2 correct.

Question Number: 26

Tier: 1 **Group:** 2

K/A: 000068 (APE 68; BW A06) Control Room Evacuation

068AA2.10; Ability to determine and interpret the following as they apply to

the Control Room Evacuation:

Source range count rate

Importance Rating: **4.2**

10 CFR Part 55: XXX

10 CFR 55.43.b **:** XXX

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 0-ONOP-105, 3-OSP-300.2

Proposed references to be provided: None

Learning Objective: PTN 6902160 OBJ. 5

Cognitive Level: Higher X _

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 9 of 33)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**(continued)**	
	D. **MAINTAIN** S/G pressure at pre-Control Room evacuation values using S/G DUMP TO ATMOSPHERE hand stations.	
14.	**CHECK** Neutron Count Rate LOWERING.	**PERFORM** the following: A. IF Reactor was SHUT DOWN prior to Control Room evacuation, THEN: (1) **CHECK** Neutron Count Rate STABLE. (2) **GO TO** Attachment 14, Step 15. B. IF Reactor subcriticality can **NOT** be confirmed, THEN **ALLOW** RCS to heat up until boration is completed in subsequent steps. Attachment 14, Step 25
15.	**PROCEED** to 3B 4KV Switchgear.	

ATTACHMENT 1
(Page 1 of 2)

ALTERNATE SHUTDOWN PANEL 3C264 INSTRUMENTATION CHECK

QA RECORD PAGE

Ref. Subsection 7.2

Description	Alternate Shutdown Panel Instrument	Value	Control Room Instrument	Value	Deviation	Max. Deviation
Instrument Air Pressure	PI-3-1439-1		PI-3-1444			7.5 psig
Gamma-Metric Source Range Counts	NI-3-6649B-3 **		NI-3-6649B-1 **			1/2 decade
Gamma-Metric Wide Range % Power	NI-3-6649B-4 **		NI-3-6649B-2 **			1/2 decade
Pressurizer Level	LI-3-462-1		LI-3-462			4 percent
RCS Pressure	PI-3-406-1		* PT-3-404			50 psig
RCS T_{cold} - Loop A (Red)	TR-3-410-1		* TE-3-410A			15°F.
RCS T_{cold} - Loop B (Blue)	TR-3-410-1		* TE-3-420A			15°F.
RCS T_{cold} - Loop C (Green)	TR-3-410-1		* TE-3-430A			15°F.
RCS T_{hot} - Loop A (Red)	TR-3-413-1		* TE-3-413A			15°F.
RCS T_{hot} - Loop B (Blue)	TR-3-413-1		* TE-3-423A			15°F.
RCS T_{hot} - Loop C (Green)	TR-3-413-1		* TE-3-433A			15°F.
Stm Gen A Level	LI-3-477-1		* LT-3-477			3 percent
Stm Gen A Pressure	PI-3-1606		* PT-3-474			30 psig
Stm Gen B Level	LI-3-487-1		* LT-3-487			3 percent
Stm Gen B Pressure	PI-3-1607		* PT-3-484			30 psig
Stm Gen C Level	LI-3-497-1		* LT-3-497			3 percent
Stm Gen C Pressure	PI-3-1608		* PT-3-494			30 psig

* SAS Channel

** Applicable when nuclear flux is within the operating range of the instrument. When out of range, the instrument should be pegged in the appropriate direction to match plant conditions (i.e., low range pegged high when unit at power).

Question # 27

Given the following:

- Unit 3 experiences a Safety Injection from 100% power.
- ANN H1/5, CHRMS HIGH RADIATION, alarms.

Which one of the following completes the statements below?

IAW ARP H1/5, the crew will check __(1)__.

The loss of CHRMS channels will require the crew to evaluate __(2)__ .

A. (1) PRMS for increased radiation levels in containment
   (2) TS 3.3.3.1, Radiation Monitoring for Plant Operations

B. (1) PRMS for increased radiation levels in containment
   (2) TS 3.3.3.3, Accident Monitoring Instrumentation

C. (1) ARMS for increased radiation levels in containment
   (2) TS 3.3.3.1, Radiation Monitoring for Plant Operations

D. (1) ARMS for increased radiation levels in containment
   (2) TS 3.3.3.3, Accident Monitoring Instrumentation

Answer Analysis

Discussion: Correct answer – D. ONOP-067 Radioactive Effluent Release requires

A. Part 1 incorrect, plausible since , PRMS is the Radiation monitoring used in containment for indications of an RCS leak. It is Particulate, Iodine, and noble gases. (PING) – Not for High Rad
Part 2 incorrect, plausible since other radiation monitoring equipment is required by this Tech Spec.

B. B1 Plausible See A1
B2 correct.

C. C1 Correct
C2 Incorrect. Plausible, See A2

D. Correct IAW ARP H1/5 and TS table 3.3-3

Question Number: 27

Tier: 1 Group: 2

K/A: **WE 16EK3.4, High Containment Radiation, RO function within the control room team in such a way that procedures are adhered to and the limitation in the facility license is maintained.**

Importance Rating: 3.0

10 CFR Part 55: 41.10

10 CFR 55.43.b n/a – RO question because question is related to the Technical Specification 3.3.3.3 LCO, above the line, (table) and the initial action of the RO in ARP H 1/5. It is also RO function to review and perform the response for all ARPs that alarm.

K/A Match: The K/A is matched as the applicant must have knowledge of ARP H 1/5 related to high radiation monitoring system in containment. Also the RO applicant must have knowledge that a minimum operable channel is required to not violate the facility license (Tech Spec 3.3.3.3)

Technical

References: ARP H 1/5, Technical Specification 3.3.3.3

Proposed references None

to be provided:

Learning Objective: PTN 6902327 OBJ. 2

Cognitive Level: Higher _

Lower X_

Question Source: New x_

Modified Bank _

	Bank _
Question History:	New
Comments:	Screenshot of development references as applicable

CAUSES:
1. High radiation in containment
2. Loss of power
3. Control module removed from racks
4. Loss of signal from detector

H1/5

CHRMS HI RADIATION

DEVICE:
- RAD-3-6311A
- RAD-3-6311B

SETPOINT: 1.3×10^4 R/HR

LOCATION: N/A

ALARM CONFIRMATION

1. **CHECK** the following:
   - Indication on Panel 3C10
   - Control module in Rack QR-81/82 for level indication and for power to each module

OPERATOR ACTIONS

1. **CHECK** ARMS channels inside containment for increased levels.
2. **PERFORM** channel check test using 3-OSP-201.1, RO Daily Logs.
3. **NOTIFY** Shift Manager.
4. **REFER TO** TS 3.3.3 for additional actions.
5. **REFER TO** 0-EPIP-20101, Duties of Emergency Coordinator.

REFERENCES: Tech Spec Section 3.3.3

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. DELETED				
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

Question # 28

Given the following:

- Unit 3 is at 75% power.

Subsequently:

- ANN F1/1, RCP MOTOR/SHAFT HI VIB, alarms.
- R-3-369 RCP Vibration Recorder indicates the following for 3B RCP:
  - Shaft vibration is 20 mils and rising slowly.
  - Motor frame vibration is 4 mils and stable.

Which one of the following completes the statement below?

The crew will __(1)__ IAW __(2)__ .

A. (1) trip the reactor then trip the 3B RCP
(2) 3-ARP-097.CR.F, Control Room Annunciator Response Panel F

B. (1) trip the reactor then trip the 3B RCP
(2) 3-ONOP-041.1, Reactor Coolant Pump Off-Normal

C. (1) perform a fast load reduction
(2) 3-GOP-100, Fast Load Reduction

D. (1) perform a fast load reduction
(2) 3-GOP-103, Power Operation to Hot Standby

Answer Analysis

A. (1) Correct, ONOP foldout page contains RCP trip criteria.
(2) Incorrect, but plausible if candidate believes ARP contains RCP trip criteria and since ARPs are entered 1st, it takes precedence.

B. (1) Correct
(2) Correct, See A(1)

C. (1) Incorrect, plausible if candidate believes reactor power must be reduced to satisfy an RPS permissive (e.g. <P8) prior to tripping an RCP to prevent a unit trip. Also plausible because this action is correct for other RCP malfunctions such as loss of seal injection and high CBO flow.
(2) Incorrect, plausible since this procedure can be used for a load reduction.

D. (1) Incorrect, see C(1)
(2) Incorrect, see C(2)

Question Number: 28

Tier: 2 **Group:** 1

K/A: 003 (SF4P RCP) Reactor Coolant Pump

003A2.02; Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP

Importance Rating: **3.7**

10 CFR Part 55: 41.5

10 CFR 55.43.b : XXX

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-ONOP-0041.1

Proposed references None

to be provided:

Learning Objective: PTN 6902205A OBJ. 7

Cognitive Level: Higher X _

Lower _

Question Source: New _

Modified Bank _

Bank X

Question History: Question used on 2016 PTN Exam, RO Question 52 with exact K/A match. That exam question was Modified to change conditions and change correct answer from PTN Bank item 69022050302.

Comments: Screenshot of development references as applicable

FOLDOUT PAGE FOR PROCEDURE 3-ONOP-041.1

1. **RCP STOPPING CRITERIA**

IF any of the following RCP limits are reached, **THEN** manually Trip the Reactor, and verify Reactor Trip using the EOP network, and then stop the affected RCP, and close PCV-3-455A, PZR Spray Valve Loop C, or PCV-3-455B, PZR Spray Valve Loop B, if applicable.

- RCP pump bearing temperature on DCS - GREATER THAN OR EQUAL TO **225°F.**
- RCP motor bearing temperature on DCS - GREATER THAN OR EQUAL TO **195°F.**
- RCP stator winding temperature on DCS - GREATER THAN OR EQUAL TO **248°F.** Note exception in Foldout Page Item 4.
- Motor frame vibration, R-3-369 (Points 1, 2, 5, 6, 9, 10) - GREATER THAN OR EQUAL TO **5 MILS**. Note exception in Foldout Page Item 4.
- RCP shaft vibration, R-3-369 (Points 3, 4, 7, 8, 11, 12) - GREATER THAN OR EQUAL TO **20 MILS**. Note exception in Foldout Page Item 4.

2. **RCP SEAL CRITERIA FOR STOPPING RCP**

IF any of the following RCP limits are reached, **THEN** manually Trip the Reactor, and verify the Reactor Tripped using the EOP network, and stop the affected RCP, Close the applicable RCP CBO Isolation Valve 303A, 303B, or 303C, and Close PCV-3-455A, PZR Spray Valve Loop C, or PCV-3-455B, PZR Spray Valve Loop B, if applicable.

- RCP CBO temperatures on DCS - GREATER THAN OR EQUAL TO **260°F**.
- RCP CBO flow exceeds **4.1 gpm**
- Any Seal Stage differential pressure exceeds **2000 psid AND** respective CBO Isolation valve (CV-3-303A, 303B or 303C) is Open

3. **FAST LOAD REDUCTION CRITERIA**

IF any of the following RCP limits are reached, **THEN** perform 3-GOP-100, Fast Load Reduction.

- RCP CBO Flow - GREATER THAN **3.7 gpm AND increasing**
- DP across any Seal Stage - GREATER THAN **1700 psid AND** respective CBO Isolation valve (CV-3-303A, 303B or 303C) is Open
- ALL of the following indications exist on the same RCP indicating a failed #3 Seal
  - RCP CBO Flow - LESS THAN **0.5 gpm**
  - RCP CBO isolation vale - OPEN
  - P3 pressure - LESS THAN **100 psig**
  - P2 pressure - GREATER THAN **1000 psig**

4. **EXCEEDING VIBRATION OR STATOR TEMPERATURE LIMITS**

- For the basis of obtaining data for startup, for balancing an RCP, or for shutdown operations; the Electrical Maintenance Supervisor or Component Engineering Supervisor may authorize continued RCP operations with vibration level or stator winding temperature above stopping criteria noted in Foldout Page Item 2. This authorization is required to be obtained prior to starting the RCP.
- When in EOP network, RCP stator winding temperature on DCS - GREATER THAN OR EQUAL TO 300°F.

5. **RCP VIBRATION ASSESSMENT CRITERIA**

IF motor frame vibration, R-3-369 (Points 1, 2, 5, 6, 9, 10), is greater than or equal to 3 mils, but less than 5 mils, **THEN** contact Engineering to evaluate the condition.

Question # 29

Given the following:

- Unit 3 is at 100% power.
- 2 Letdown orifices are in service.

Subsequently:

- PCV-3-145, Low Pressure Letdown Controller, fails OPEN.
- Letdown flow rises to 131 gpm.

Which of the following completes the statements below?

TCV-3-143, L/D Demineralizer Divert Valve, __(1)__ to VCT in order to prevent __(2)__.

A. (1) will AUTOMATICALLY align
   (2) CVCS Demin resin carry over to the RCS filters

B. (1) will AUTOMATICALLY align
   (2) Channeling of the CVCS Demins

C. (1) is required to be MANUALLY aligned
   (2) CVCS Demin resin carry over to the RCS filters

D. (1) is required to be MANUALLY aligned
   (2) Channeling of the CVCS Demins

Answer Analysis

A Incorrect.
Part 1 incorrect, plausible since this is the addition of the flow obtained between a 45 gpm and a 60 gpm orifices.
Part 2 incorrect, plausible since, demin resin will deposit in the RCS filters.

B Incorrect.
Part 1 incorrect, plausible since this is the addition of the flow obtained between a 45 gpm and a 60 gpm orifices.
Part 2 correct.

C Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since, demin resin will deposit in the RCS filters.

D Correct. 3-OP-047, directs to limit letdown flow to below 120 gpm to prevent channeling in the demins.

Question Number: 29

Tier: 2 Group 1

K/A: 004K6.22, 2.6**, 41.7 Knowledge of the effect of a malfunction on the CVCS systems:** Design minimum and maximum flow rates for letdown system

:

Importance Rating: **2.6; 10 CFR Part 55:** 41.7

10 CFR 55.43.b **:**N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, identify the implication of annunciators alarmed and not alarmed and determine IF operator action is required to protect the Letdown demineralizers from the maximum design flow rate

SRO Justification: N/A

Technical References: ARP A 5/5 CVCS LP LTDN LINE HI FLOW/PRESS

Proposed references None
Learning Objective: PTN 6902113 OBJ. 6

Cognitive Level: Higher X_

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

CAUSES:
1. Hi letdown flow rate (too many orifices in service)
2. PCV-3-145 control failure

A5/5

CVCS
LP LTDN LINE
HI FLOW/PRESS

DEVICE:	SETPOINT:	LOCATION:
• PT-3-145	490 psig	N/A
• FE-3-150	130 gpm	N/A

ALARM CONFIRMATION

1. **CHECK** PI-3-145, LOW PRESS LTDN PRESS greater than 490 psig on VPA.
2. **CHECK** FI-3-150, LOW PRESS LTDN FLOW greater than 130 gpm on VPA.

OPERATOR ACTIONS

NOTE

Demineralizer is rated for 120 gpm.

1. IF HI flow, THEN:
   A. **PLACE** TCV-3-143, L/D DEMINERALIZER DIVERT VALVE to the DIVERT position to bypass the Demineralizers.
   B. **REMOVE** orifices from service to reduce flow.
   C. **CHECK** PCV-3-145, LOW PRESSURE LTDN CONTROLLER operating properly.
2. IF HI pressure, THEN:
   A. **CHECK** PCV-3-145, LOW PRESSURE LTDN CONTROLLER operating properly.
   B. IF PCV-3-145, LOW PRESSURE LTDN CONTROLLER is **NOT** operating properly, THEN **PERFORM** one of the following:
      (1) **TAKE** manual control at the Letdown Controller station.
      (2) **CONTROL** letdown pressure locally as follows:
         a. **THROTTLE** open 3-309C, BYPASS VALVE to reduce pressure.
         b. **CLOSE** 3-309A, NRHX L/D OUTLET PCV-3-145 INLET ISOL, to isolate PCV-3-145, LOW PRESSURE LTDN CONTROLLER.

REFERENCES: FPL Drawing 5613-M-3047, Sheet 1, CVCS – Charging and Letdown

Letdown

- Three parallel flow-limiting orifices located after RHX
  - One orifice rated at 45 gpm [CV-3(4)-200A] and the other two at 60 gpm each [CV-3(4)-200B/C], under normal RCS pressure
  - Typically, CV-3(4)-200A is aligned, with CV-3(4)-200B and CV-3(4)-200C in standby
  - Any two orifices may be aligned at a given time to raise letdown flow (e.g., when RCS pressure is lower than normal or elevated letdown flow is required for purification or to compensate for RCS expansion during heatup); however, the maximum allowable letdown flow is 120 gpm, to minimize the possibility of channeling in the demineralizers

Objectives 3, 4 & 5

Question # 30

Given the following:

- Unit 4 is in MODE 5.
- The PZR is solid.
- OMS in low pressure operation.
- 4B RHR loop is in operation.
- PCV-4-145, Low Pressure Letdown Controller, is in MANUAL due to a failure of the control system.
- Two CCW heat exchangers are in service.

Subsequently:

- The crew places another CCW heat exchanger in operation.

Which one of the following completes the statement below?

In order to maintain RCS pressure stable, the operator will operate __(1)__.

A. PCV-4-145, by pressing the DOWN arrow

B. PCV-4-145, by pressing the UP arrow

C. FCV-4-605, RHR HX bypass, by turning the setpoint potentiometer clockwise

D. FCV-4-605, RHR HX bypass, by turning the setpoint potentiometer counter-clockwise

Answer Analysis

Placing another CCW HX in operation will cause a cooldown, which will make the RCS pressure drop. PCV-4-145 is used to control RCS pressure in solid plant conditions and is a reverse-acting controller.

A. Correct valve, incorrect direction. Plausible if the applicant thinks that it is a direct-acting controller.

B. Correct.

C. Incorrect valve. Plausible if the applicant thinks that opening this valve will decrease the effectiveness of the RHR HX by robing flow from it.

D. Incorrect valve. Plausible if the applicant thinks that closing this valve will decrease the effectiveness of the RHR HX by throttling flow to it.

Question Number: 30

Tier: 2 **Group:** 1

K/A: 004 (SF1; SF2 CVCS) Chemical and Volume Control

004A1.03; Ability to predict and/or monitor changes in parameters

(to prevent exceeding design limits) associated with operating the CVCS controls including:

RCS pressure

Importance Rating: **3.8**

10 CFR Part 55: 41.5

10 CFR 55.43.b : XXX

K/A Match:

SRO Justification: N/A

Technical References: 5613-M-3047 SH 1

Proposed references to be provided: None

Learning Objective: PTN 6902113 OBJ. 6

Cognitive Level: Higher _

Lower _

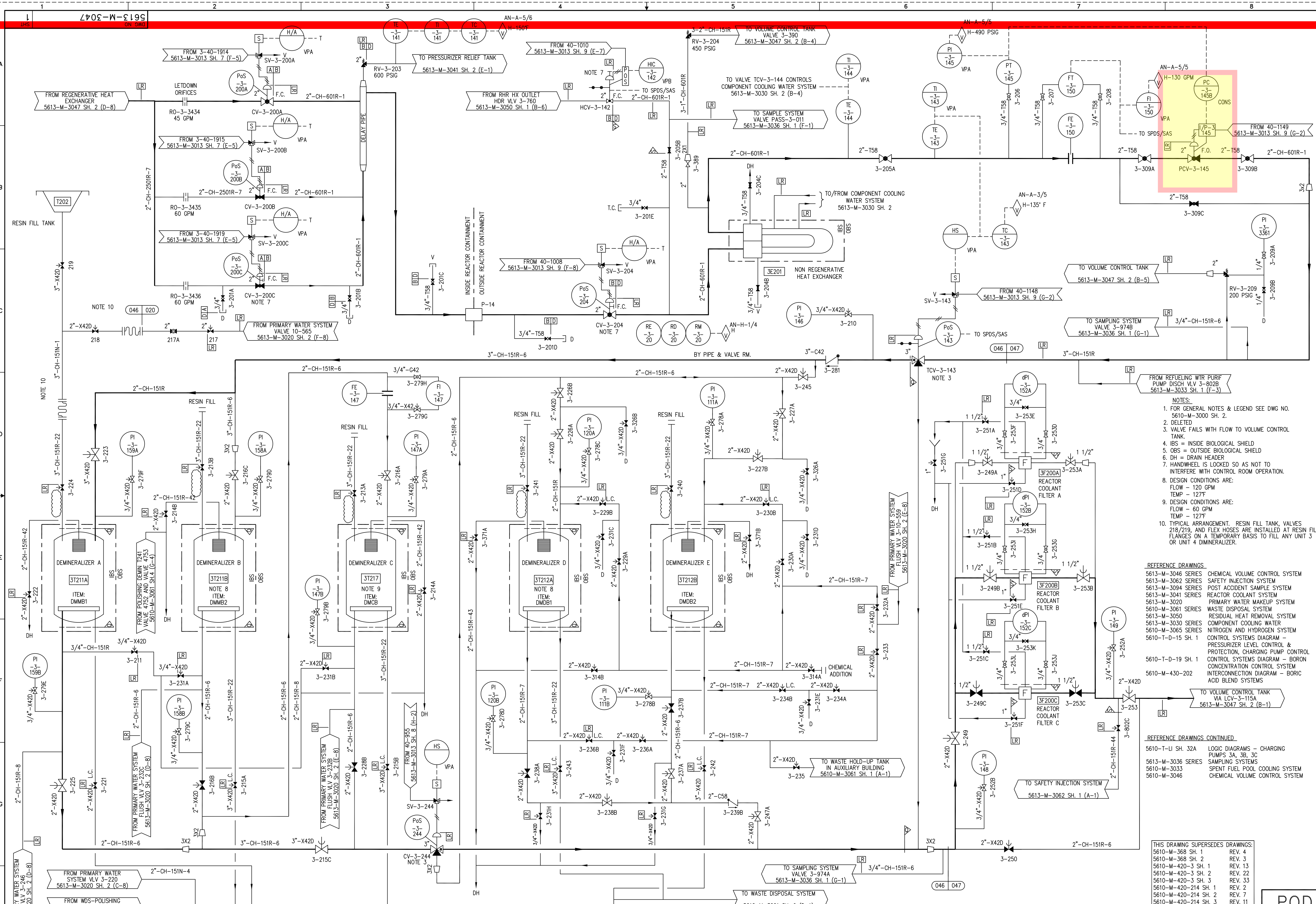
Question Source: New X

Modified Bank _

Bank

Question History: New

Comments:



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2.
2. DELETED
3. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK.
4. IBS = INSIDE BIOLOGICAL SHIELD
5. OBS = OUTSIDE BIOLOGICAL SHIELD
6. DH = DRAIN HEADER
7. HANDWHEEL IS LOCKED SO AS NOT TO INTERFERE WITH CONTROL ROOM OPERATION.
8. DESIGN CONDITIONS ARE:
   FLOW – 120 GPM
   TEMP – 127°F
9. DESIGN CONDITIONS ARE:
   FLOW – 60 GPM
   TEMP – 127°F
10. TYPICAL ARRANGEMENT. RESIN FILL TANK, VALVES 218/219, AND FLEX HOSES ARE INSTALLED AT RESIN FILL FLANGES ON A TEMPORARY BASIS TO FILL ANY UNIT 3 OR UNIT 4 DIMINERALIZER.

REFERENCE DRAWINGS

5613-M-3046 SERIES	CHEMICAL VOLUME CONTROL SYSTEM
5613-M-3062 SERIES	SAFETY INJECTION SYSTEM
5613-M-3094 SERIES	POST ACCIDENT SAMPLE SYSTEM
5613-M-3041 SERIES	REACTOR COOLANT SYSTEM
5613-M-3020	PRIMARY WATER MAKEUP SYSTEM
5610-M-3061 SERIES	WASTE DISPOSAL SYSTEM
5613-M-3050	RESIDUAL HEAT REMOVAL SYSTEM
5613-M-3030 SERIES	COMPONENT COOLING WATER
5610-M-3065 SERIES	NITROGEN AND HYDROGEN SYSTEM
5610-T-D-15 SH. 1	CONTROL SYSTEMS DIAGRAM – PRESSURIZER LEVEL CONTROL & PROTECTION, CHARGING PUMP CONTROL
5610-T-D-19 SH. 1	CONTROL SYSTEMS DIAGRAM – BORON CONCENTRATION CONTROL SYSTEM
5610-M-430-202	INTERCONNECTION DIAGRAM – BORIC ACID BLEND SYSTEMS

REFERENCE DRAWINGS CONTINUED

5610-T-LI SH. 32A	LOGIC DIAGRAMS – CHARGING PUMPS 3A, 3B, 3C
5613-M-3036 SERIES	SAMPLING SYSTEMS
5610-M-3033	SPENT FUEL POOL COOLING SYSTEM
5610-M-3046	CHEMICAL VOLUME CONTROL SYSTEM

THIS DRAWING SUPERSEDES DRAWINGS:

5610-M-368 SH. 1	REV. 4
5610-M-368 SH. 2	REV. 3
5610-M-420-3 SH. 1	REV. 13
5610-M-420-3 SH. 2	REV. 22
5610-M-420-3 SH. 3	REV. 33
5610-M-420-214 SH. 1	REV. 2
5610-M-420-214 SH. 2	REV. 7
5610-M-420-214 SH. 3	REV. 11

POD

NOTE: THIS DWG IS MADE FROM:

FPL POD	5610-T-E-4505 SH. 1	REV. 84
FPL POD	5610-T-E-4505 SH. 2	REV. 55
FPL POD	5610-T-E-4505 SH. 5	REV. 7
FPL POD	5610-T-E-4515 SH. 2	REV. 22

REV	DATE	REVISION	BY	CH	APP	APP
16	3-6-00	ISSUED AS-BUILT PER CRN-M-9848 (PC/M 99-046), CRN-M-9991 (PC/M 99-061) AND CRN-M-9993 (PC/M 99-061).	JPG	RH	BSG	JMM
23	8/21/18	ISSUED AS-BUILT PER EC 288195.	RH	AFG	–	BB
22	5/23/12	ISSUED AS-BUILT PER EC 274476.	RV	RH	–	MA
21	9/21/09	ISSUED AS-BUILT PER PC/M 06-103.	RV	RH	BB	SB
20	6/1/07	ISSUED AS-BUILT PER CRN M-11762 (PC/M 03-106)	RH	RV	PJV	RSV

REV	DATE	REVISION	BY	CH	APP	APP
19	3/8/05	ISSUED AS-BUILT PER CRN-M-11262 (PC/M 03-106).	RH	RRW	BSG	JTL
18	10/15/04	ISSUED AS-BUILT PER PC/M 03-132.	RH	SB	PRB	JTL
17	6-17-02	ISSUED AS-BUILT PER CRN-M-10465 (PC/M 00-016).	RH	RV	AGM	JTL
0	3-5-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-045.	MAB	MD		LRB

FPL

TURKEY POINT NUCLEAR UNIT 3

P & ID

CHEMICAL AND VOLUME CONTROL SYSTEM
CHARGING AND LETDOWN

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3047 | SYS 047 | REV 23 | SHEET 1

8 5613-M-3047_SH 1.DWG

Question # 31

Given the following:

- Unit 4 is in MODE 4.
- 4B RHR loop is providing RCS cooling.

Subsequently:

- A Unit 4 surveillance error initiates an inadvertent Safety Injection.

Which one of the following completes the statements below?

FCV-4-605, RHR Heat Exchanger Bypass Valve, __(1)__ receive a signal from the safeguards cabinets to fully CLOSE.

MOV-4-862A & MOV-4-862B, RHR suction valves from the RWST __(2)__ AUTOMATICALLY open.

A. (1) will NOT
(2) will

B. (1) will NOT
(2) will NOT

C. (1) will
(2) will

D. (1) will
(2) will NOT

Answer Analysis

A. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since, these valves are part of the SI injection flowpath.

B. Correct.
FCV-4-605, does not receive a closure signal from the safeguards cabinets.
RHR suction valves MOV 862A/863A (RWST) are deenergized and do not receive an SI signal.

C. Incorrect.
Part 1 incorrect, plausible since, other plant equipment is AUTOMATICALLY operated by the safeguards system in this fashion in order to maximize cooling flow to the plant, i.e. POV-4-4883(4), ICW to TPCW Isolations, go close on SI.
Part 2 incorrect, plausible since, these valves are part of the SI injection flowpath.

D. Incorrect.
Part 1 incorrect, plausible since, other plant equipment is AUTOMATICALLY operated by the safeguards system in this fashion in order to maximize cooling flow to the plant, i.e. POV-4-4883(4), ICW to TPCW Isolations, go close on SI.
Part 2 correct.

Question Number: 31

Tier: 2 **Group:** 1

K/A: 005K1.06

Knowledge of the physical connections and/or cause effect relationships between the RHRS and ECCS

Importance Rating: **3.5**

10 CFR Part 55: 41.2

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions and identify automatic system response (cause and effect of RHR with ECCS SI signal) for the RHR system event with an inadvertent ECCS actuation.

SRO Justification: N/A

Technical References: 5614-M-3050 SH1

Proposed references to be provided: None

Learning Objective: PTN 692327 OBJ. 10

Cognitive Level: Higher X

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

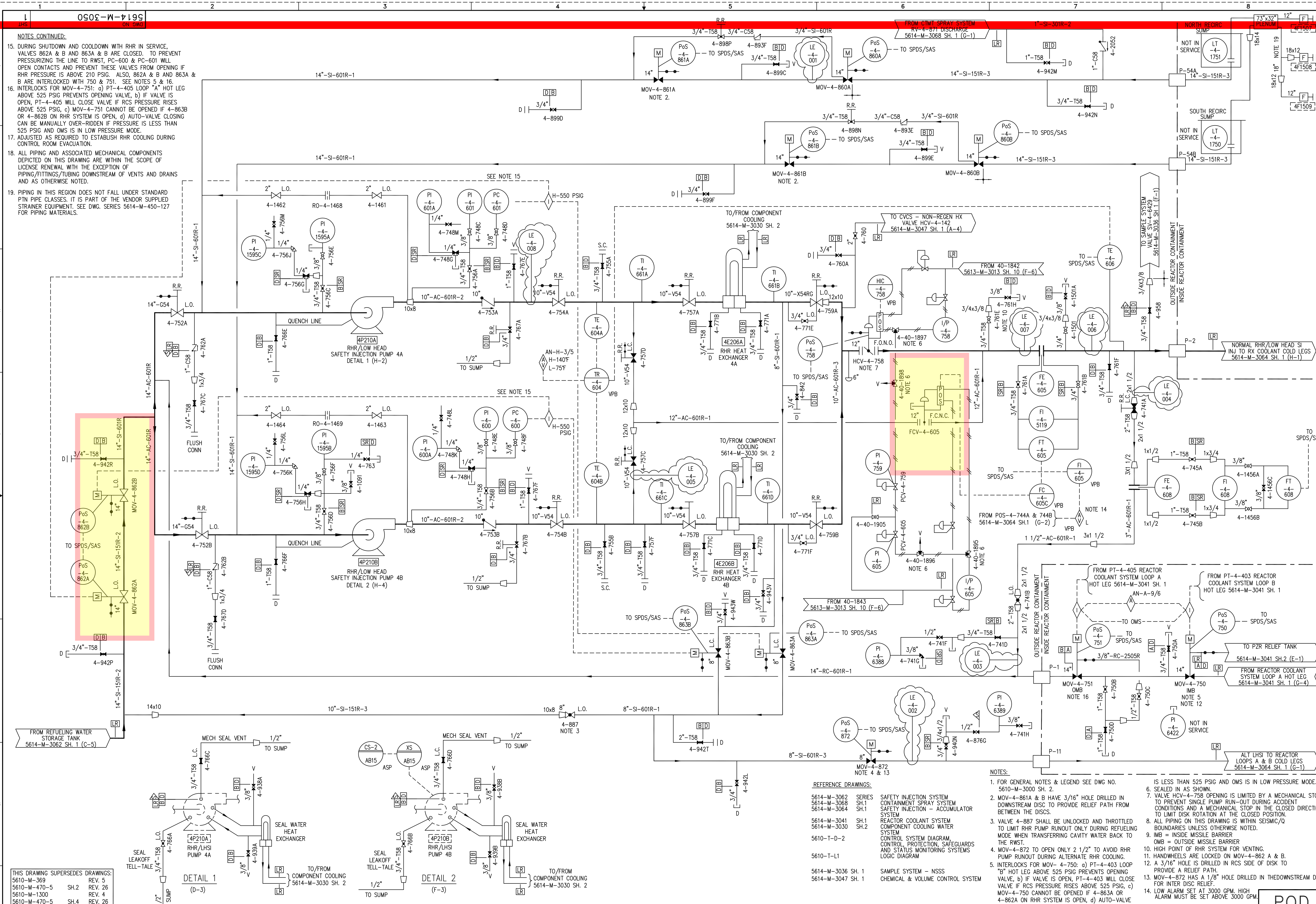
MOTOR OPERATED VALVES

VALVE NO.	VALVE DESCRIPTION	BKR NO.	CONTROL SWITCH & INDICATING LIGHTS	AUTO ACTION	LIGHT PANEL
744A & B	LOW Head SI RHR to Loops A B, & C	3[4] 0722 3[4] 0613	VPB	Opens on "S" signal	SI Panel
750 & 751	Normal RHR suction from Loop C [A] hot leg	3[4] 0615 3[4] 0731	VPB	Closes if RCS pressure 525±10 PSIG (Interlock 862A&B, 863A&B)	--
750 & 751	Over-ride Pushbutton		VPB	Manual - Stops Closing Sequence	
843A & B	HHSI to cold legs	3[4] 0738 3[4] 0622	VPB	Opens on "S" signal	SI Panel
860A & B	Recirc. sump to RHR pump suction	3[4] 0711 3[4] 0604	VPB	None --	
861A & B	Recirc. sump to	3[4] 0728	VPB	None	--

	RHR pump suction	3[4] 0603			
862A & B	RHR pumps suction from RWST	3[4] 0720 3[4] 0616	VPB	Interlock with PC-600 & 601 RCS >210 PSIG prevents opening	--

AIR OPERATED VALVES

VALVE	DESCRIPTION	CONTROL FUNCTION	NORMAL/FAIL POSITION
605	RHR HX Bypass	Throttle-Maintain RHR system flow constant	NC/FC
758	RHR HX Outlet Common Header	Throttle-HX flow rate	NO/FO



NOTES CONTINUED:

15. DURING SHUTDOWN AND COOLDOWN WITH RHR IN SERVICE, VALVES 862A & B AND 863A & B ARE CLOSED. TO PREVENT PRESSURIZING THE LINE TO RWST, PC-600 & PC-601 WILL OPEN CONTACTS AND PREVENT THESE VALVES FROM OPENING IF RHR PRESSURE IS ABOVE 210 PSIG. ALSO, 862A & B AND 863A & B ARE INTERLOCKED WITH 750 & 751. SEE NOTES 5 & 16.
16. INTERLOCKS FOR MOV-4-751: a) PT-4-405 LOOP "A" HOT LEG ABOVE 525 PSIG PREVENTS OPENING VALVE, b) IF VALVE IS OPEN, PT-4-405 WILL CLOSE VALVE IF RCS PRESSURE RISES ABOVE 525 PSIG, c) MOV-4-751 CANNOT BE OPENED IF 4-863B OR 4-862B ON RHR SYSTEM IS OPEN, d) AUTO-VALVE CLOSING CAN BE MANUALLY OVER-RIDDEN IF PRESSURE IS LESS THAN 525 PSIG AND OMS IS IN LOW PRESSURE MODE.
17. ADJUSTED AS REQUIRED TO ESTABLISH RHR COOLING DURING CONTROL ROOM EVACUATION.
18. ALL PIPING AND ASSOCIATED MECHANICAL COMPONENTS DEPICTED ON THIS DRAWING ARE WITHIN THE SCOPE OF LICENSE RENEWAL WITH THE EXCEPTION OF PIPING/FITTINGS/TUBING DOWNSTREAM OF VENTS AND DRAINS AND AS OTHERWISE NOTED.
19. PIPING IN THIS REGION DOES NOT FALL UNDER STANDARD PTN PIPE CLASSES. IT IS PART OF THE VENDOR SUPPLIED STRAINER EQUIPMENT. SEE DWG. SERIES 5614-M-450-127 FOR PIPING MATERIALS.

REFERENCE DRAWINGS:

5614-M-3062	SERIES	SAFETY INJECTION SYSTEM
5614-M-3068	SH.1	CONTAINMENT SPRAY SYSTEM
5614-M-3064	SH.1	SAFETY INJECTION – ACCUMULATOR SYSTEM
5614-M-3041	SH.1	REACTOR COOLANT SYSTEM
5614-M-3030	SH.2	COMPONENT COOLING WATER SYSTEM
5610-T-D-2		CONTROL SYSTEM DIAGRAM, CONTROL, PROTECTION, SAFEGUARDS AND STATUS MONITORING SYSTEMS
5610-T-L1		LOGIC DIAGRAM
5614-M-3036	SH. 1	SAMPLE SYSTEM – NSSS
5614-M-3047	SH. 1	CHEMICAL & VOLUME CONTROL SYSTEM

NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2.
2. MOV-4-861A & B HAVE 3/16" HOLE DRILLED IN DOWNSTREAM DISC TO PROVIDE RELIEF PATH FROM BETWEEN THE DISCS.
3. VALVE 4-887 SHALL BE UNLOCKED AND THROTTLED TO LIMIT RHR PUMP RUNOUT ONLY DURING REFUELING MODE WHEN TRANSFERRING CAVITY WATER BACK TO THE RWST.
4. MOV-4-872 TO OPEN ONLY 2 1/2" TO AVOID RHR PUMP RUNOUT DURING ALTERNATE RHR COOLING.
5. INTERLOCKS FOR MOV- 4-750: a) PT-4-403 LOOP "B" HOT LEG ABOVE 525 PSIG PREVENTS OPENING VALVE, b) IF VALVE IS OPEN, PT-4-403 WILL CLOSE VALVE IF RCS PRESSURE RISES ABOVE 525 PSIG, c) MOV-4-750 CANNOT BE OPENED IF 4-863A OR 4-862A ON RHR SYSTEM IS OPEN, d) AUTO-VALVE CLOSING CAN BE MANUALLY OVER-RIDDEN IF PRESSURE IS LESS THAN 525 PSIG AND OMS IS IN LOW PRESSURE MODE.
6. SEALED IN AS SHOWN.
7. VALVE HCV-4-758 OPENING IS LIMITED BY A MECHANICAL STOP TO PREVENT SINGLE PUMP RUN-OUT DURING ACCIDENT CONDITIONS AND A MECHANICAL STOP IN THE CLOSED DIRECTION TO LIMIT DISK ROTATION AT THE CLOSED POSITION.
8. ALL PIPING ON THIS DRAWING IS WITHIN SEISMIC/Q BOUNDARIES UNLESS OTHERWISE NOTED.
9. IMB = INSIDE MISSILE BARRIER
   OMB = OUTSIDE MISSILE BARRIER
10. HIGH POINT OF RHR SYSTEM FOR VENTING.
11. HANDWHEELS ARE LOCKED ON MOV-4-862 A & B.
12. A 3/16" HOLE IS DRILLED IN RCS SIDE OF DISK TO PROVIDE A RELIEF PATH.
13. MOV-4-872 HAS A 1/8" HOLE DRILLED IN THEDOWNSTREAM DISC FOR INTER DISC RELIEF.
14. LOW ALARM SET AT 3000 GPM. HIGH ALARM MUST BE SET ABOVE 3000 GPM.

NOTE CONTINUED @ UPPER LEFT CORNER.

POD

THIS DRAWING SUPERSEDES DRAWINGS:

5610-M-369		REV. 5
5610-M-470-5	SH.2	REV. 26
5610-M-1300		REV. 4
5610-M-470-5	SH.4	REV. 26
5610-M-370		REV. 3
5610-M-450-53		REV. 22
5610-M-470-5	SH.3	REV. 31
5610-M-470-45	SH.2	REV. 6
5610-M-470-45	SH.3	REV. 10
5610-M-470-45	SH.4	REV. 4

NOTE: THIS DWG IS MADE FROM:

FPL POD	5610-T-E-4510 SH.1	REV. 101
FPL POD	5610-T-E-4510 SH.2	REV. 72
FPL POD	5610-T-E-4515 SH.2	REV. 22

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
37	11-25-09	ISSUED AS-BUILT PER CRN M-12954 (PC/M 08-067).	RH	BB	PRB	EB	31	6-20-07	ISSUED AS-BUILT PER CRN M-12028 (PC/M 07-021).	RH	RV	BSG	ASD
36	11/23/09	ISSUED AS-BUILT PER CRN M-12949 (PC/M 08-067).	RH	RV	DW	JEA	30	1-19-07	ISSUED AS-BUILT PER CRN-M-11923 (PC/M 06-073).	RV	BB	–	PRB
35	11/11/09	ISSUED AS-BUILT PER CRN-M-12828 (PC/M 08-067).	RV	RH	ABR	PRB	39	2-17-15	ISSUED AS-BUILT PER EC 242426; INCORP. CRN-001.	RH	CW	–	SB
34	6/12/09	ISSUED AS-BUILT PER CRN-M-12754 (PC/M 09-034).	RV	BB	PJV	PRB	38	2-17-10	ISSUED AS-BUILT PER PC/M 04-104.	RH	BB	SAG	VMG
33	1/17/09	ISSUED AS-BUILT PER PC/M 04-104. (PARTIAL)	RV	SB	SB	PRB	0	4-2-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	TAM	MD	RAS/SVM	LRB
32	5-2-08	ISSUED AS-BUILT PER PC/M 06-031 AND INCORP. CRN-M-12309.	RH	BB	GCV	PRB	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-044.	—	—	—	—

FPL

TURKEY POINT NUCLEAR UNIT 4

P & I D

RESIDUAL HEAT REMOVAL SYSTEM

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5614-M-3050

SYS 050

SHEET 1

REV 39

Question # 32

Given the following:

- Unit 3 is at 100% power.

Subsequently:

- A CCW rupture has occurred.
- FI-3-613A, Flow Ind For CCW Loop A, indicates 0 gpm.
- FI-3-613B, Flow Ind For CCW Loop B, indicates 0 gpm.

Which one of the following completes the statements below?

The operating charging pump is required to be run at __(1)__ speed until alternate cooling from the __(2)__ System is established.

A. (1) MAXIMUM
(2) Unit 4 CCW

B. (1) MAXIMUM
(2) Service Water

C. (1) 50%
(2) Unit 4 CCW

D. (1) 50%
(2) Service Water

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. (1) Correct, as directed by 3-ONOP-030
(2) Incorrect, plausible since there are installed unit cross connect valves for the HHSI pumps

B. (1) Correct
(2) Correct

C. (1) Incorrect, plausible to assume slower pump speed will minimize heat input
(2) Incorrect, see A(2)

D. (1) Incorrect, see C(1)
(2) Correct

Question Number: 32

Tier: 2 **Group:** 1

K/A: 006 (SF2; SF3 ECCS) Emergency Core Cooling

006K4.01; Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following:

Cooling of centrifugal pump bearings

Importance Rating: **2.6**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** XXX

K/A Match: I consider the K/A matched even though the charging pumps are not centrifugal pumps. The fact that they are PDPs does not change the effect of loss of CCW on cooling.

SRO Justification: N/A

Technical References: 3-ONOP-030

Proposed references to be provided: None

Learning Objective: PTN 6902229 OBJ. 4

Cognitive Level: Higher _

Lower X _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

3.2 Subsequent Operator Actions (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **CHECK** flow normal in both Component Cooling Water headers. • FI-3-613A, FLOW IND FOR CCW LOOP A • FI-3-613B, FLOW IND FOR CCW LOOP B	IF CCW flow to RCPs can **NOT** be established, THEN: **A.** Manually **TRIP** the reactor. **B.** **ENSURE** reactor trip per EOP. **C.** **STOP** all RCPs. **D.** **ISOLATE** Letdown and Excess Letdown. **E.** IF any Charging Pump is operating, THEN **OPERATE** at maximum speed until Attachment 1, Control of Emergency Cooling Water to Charging Pumps is COMPLETE. **F.** **ESTABLISH** emergency cooling water to desired Charging Pump(s) per Attachment 1, Control of Emergency Cooling Water to Charging Pumps.

ATTACHMENT 1

Control of Emergency Cooling Water to Charging Pumps

(Page 1 of 6)

NOTE

- Emergency cooling water supply hose has a quick disconnect fitting on one end and a cam lock fitting on the other end.
- Loss of off-site power in coincidence with a loss of CCW will require the diesel driven Service Water Pump to be in service in order to provide emergency cooling water to the Charging Pumps.

1. **CONNECT** cam lock fitting end of emergency cooling water supply hose to 3-70-179A, SERVICE WATER CONNECTION INSIDE UNIT 3 CHARGING PUMP ROOM.

2. **CONSULT** with Unit 3 Reactor Operator to determine desired charging pump.

3. **ENSURE** desired Charging Pump is STOPPED OR running at maximum speed.

4. **CONNECT** quick disconnect fitting end of emergency cooling water supply hose to emergency hose connection on desired Charging Pump:

   * 3-10-291, EMERGENCY HOSE CONNECTION TO CHARGING PUMP A OIL COOLER SUPPLY

   * 3-10-289, EMERGENCY HOSE CONNECTION TO CHARGING PUMP B OIL COOLER SUPPLY

   * 3-10-299, EMERGENCY HOSE CONNECTION TO CHARGING PUMP C OIL COOLER SUPPLY

Question # 33

Given the following:

- Unit 3 experiences a steam generator tube rupture.
- The crew is commencing the RCS depressurization IAW 3-EOP-E-3, Steam Generator Tube Rupture.
- QSPDS displays for CET subcooling indication is NOT available.
- Core Exit TCs are reading 505°F and stable.

Which one of the following completes the statements below?

The LOWEST pressure the RCS can be depressurized to while maintaining the MINIMUM required CET Subcooling is ______.

A. 585 psig

B. 600 psig

C. 830 psig

D. 845 psig

Answer Analysis

A. Incorrect, plausible since this response is obtained if 19°F is subtracted instead of added to 505°F.

B. Incorrect, plausible since this response is obtained if 19°F is subtracted instead of added to 505°F and the conversion from psia to psig is not carried out.

C. Correct. 505°F + 19°F= 524°F. Saturation pressure for 524°F is 824 psig. 830 psig is the lowest pressure that does not exceed 524°F.

D. Incorrect, plausible since this is the psia number for the correct response.

Question Number: 33

Tier: 2 **Group:** 1

K/A: 006K6.18

Knowledge of the effect or loss of Subcooling Monitoring indicators on the ECCS

Importance Rating: **3.5**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions and determine subcooling using steam tables once indication is lost. The effect on ECCS is that the manual subcooling results are now used to determine the pressure to secure max spray and if SI (ECCS injection) flow can be reduced and terminated. The referenced E-3 step is leading to reducing ECCS injection flow to minimize the RCS leak and requires using the correct Subcooling indication.

SRO Justification: N/A

Technical References: Tech Specs 3.3.3

UFSAR Section, Instrumentation and Controls

Lesson Plan 6902171, QSPDS

E-3, SGTR

Proposed references to be provided: None

Learning Objective: PTN 6902171 OBJ. 5 & 6

Cognitive Level: Higher X

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

17. Depressurize RCS To Minimize Break Flow And Refill PRZ

a. Normal PRZ Spray – AVAILABLE

a. Observe CAUTIONS and NOTE prior to Step 18 and go to Step 18.

b. Spray PRZ with maximum available spray until any of the following conditions satisfied using Attachment 6, Section 1.0, as a reference:

* Both of the following:
  - RCS pressure – LESS THAN RUPTURED S/G(s) PRESSURE
  - PRZ level – GREATER THAN 7%[48%]

  OR

* Both of the following:
  - RCS pressure – WITHIN 300 PSI OF RUPTURED S/G(s) PRESSURE
  - PRZ level – GREATER THAN 37%[50%]

  OR

* PRZ level – GREATER THAN 73%[60%]

  OR

* RCS subcooling based on Core Exit TCs – LESS THAN 19°F[73°F]

Question # 34

Given the following:

- Unit 3 is MODE 4.
- RHR cooling is in service.
- OMS is in Low Pressure Ops.

Subsequently:

- PRT level, pressure and temperature begin to rise
- Unit 3 Pressurizer Safety Valve Acoustic Monitoring System lights all remain DARK.

Which one of the following completes the statements below?

__(1)__ actuation is the cause of the PRT pressure increase.

If the source of the PRT pressure increase is not isolated, the __(2)__ will AUTOMATICALLY protect the tank.

A. (1) RV-3-706, RHR pump discharge header relief,
(2) PRT to waste gas relief valve, 3-516A

B. (1) OMS
(2) PRT rupture disc

C. (1) RV-3-706, RHR pump discharge header relief,
(2) PRT rupture disc

D. (1) OMS
(2) PRT to waste gas relief valve, 3-516A

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice. PZR PORVs, safeties, and the RHR discharge header relief go to the PRT. OMS controls the PORVs. Without the acoustic lights lit, the PORVs did not actuate open. The waste gas line relief is not sized to protect the PRT

A. (1) Correct,
(2) Incorrect, Plausible to believe that RV-3-516A could automatically protect the PRT. However, this valve is very small (3/8 inch) and the line also has a control / isolation valve .

B. (1) Incorrect, Plausible as OMS is designed to operate the PZR PORVs However with the Accoustic lights dark, the PORVs did not actuate or open
(2) Correct

C. Correct. PRT pressure will increase with a lifting RV-706. .

D. (1) Incorrect, See B1
(2) Incorrect. See A2

Question Number: 34

Tier: 2 **Group:** 1

K/A: 007 (SF5 PRTS) Pressurizer Relief/Quench Tank

007A3.01; Ability to monitor automatic operation of the PRTS, including:

Components which discharge to the PRT

Importance Rating: **2.7**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** XXX

K/A Match: The K/A is matched as the applicant must recognize that RV-706 goes to the PRT and that the rupture discs are the protection device that will limit pressure to 100#s

SRO Justification: N/A

Technical References: 5614-M-3041 SH 2

Proposed references to be provided: None

Learning Objective: PTN 6902107 OBJ. 5 & 6

Cognitive Level: Higher _

Lower X_

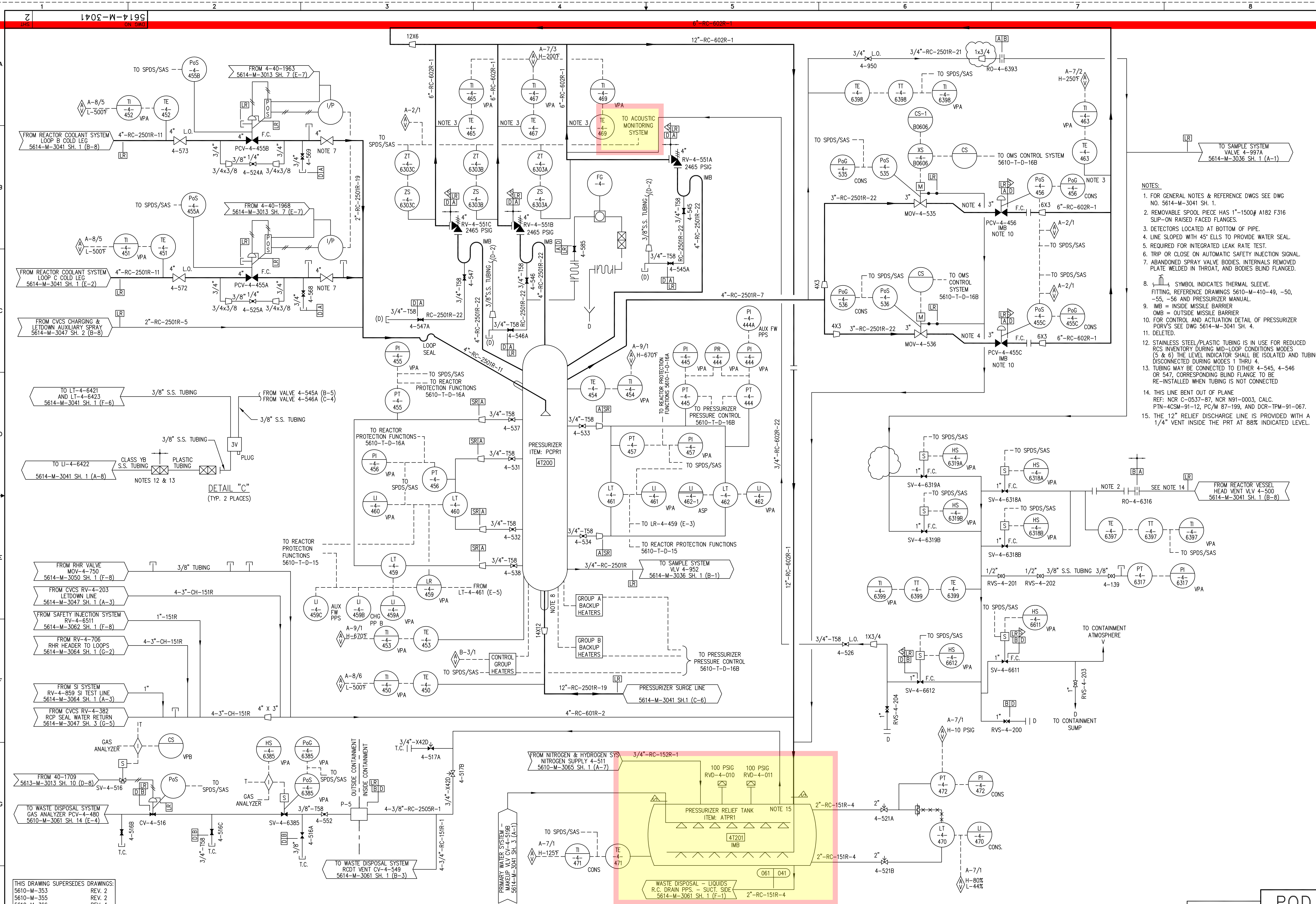
Question Source: New _

Modified Bank _

Bank X

Question History: This question was used on the PTN 2015 NRC exam. RO question #6

Comments: Screenshot of development references as applicable



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5614-M-3041 SH. 1.
2. REMOVABLE SPOOL PIECE HAS 1"-1500# A182 F316 SLIP-ON RAISED FACED FLANGES.
3. DETECTORS LOCATED AT BOTTOM OF PIPE.
4. LINE SLOPED WITH 45° ELLS TO PROVIDE WATER SEAL.
5. REQUIRED FOR INTEGRATED LEAK RATE TEST.
6. TRIP OR CLOSE ON AUTOMATIC SAFETY INJECTION SIGNAL.
7. ABANDONED SPRAY VALVE BODIES. INTERNALS REMOVED PLATE WELDED IN THROAT, AND BODIES BLIND FLANGED.
8. SYMBOL INDICATES THERMAL SLEEVE. FITTING, REFERENCE DRAWINGS 5610-M-410-49, -50, -55, -56 AND PRESSURIZER MANUAL.
9. IMB = INSIDE MISSILE BARRIER OMB = OUTSIDE MISSILE BARRIER
10. FOR CONTROL AND ACTUATION DETAIL OF PRESSURIZER PORV'S SEE DWG 5614-M-3041 SH. 4.
11. DELETED.
12. STAINLESS STEEL/PLASTIC TUBING IS IN USE FOR REDUCED RCS INVENTORY DURING MID-LOOP CONDITIONS MODES (5 & 6) THE LEVEL INDICATOR SHALL BE ISOLATED AND TUBING DISCONNECTED DURING MODES 1 THRU 4.
13. TUBING MAY BE CONNECTED TO EITHER 4-545, 4-546 OR 547, CORRESPONDING BLIND FLANGE TO BE RE-INSTALLED WHEN TUBING IS NOT CONNECTED
14. THIS LINE BENT OUT OF PLANE REF: NCR C-0537-87, NCR N91-0003, CALC. PTN-4CSM-91-12, PC/M 87-199, AND DCR-TPM-91-067.
15. THE 12" RELIEF DISCHARGE LINE IS PROVIDED WITH A 1/4" VENT INSIDE THE PRT AT 88% INDICATED LEVEL.

THIS DRAWING SUPERSEDES DRAWINGS:	
5610-M-353	REV. 2
5610-M-355	REV. 2
5610-M-366	REV. 4
5610-M-367	REV. 2
5610-M-410-91 SH. 1	REV. 13
5610-M-410-91 SH. 2	REV. 30
5610-M-410-202 SH. 1	REV. 4
5610-M-410-202 SH. 2	REV. 6
5610-M-1288	REV. 4
5610-M-1310	REV. 2

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
32	4-27-05	ISSUED AS-BUILT PER PC/M 04-065 & INCORP. CRN-M-11240.	RH	RV	BSG	JTL	38	2/6/13	ISSUED AS-BUILT PER EC 247470 (ITOP 13-02-027).	RH	AFG	–	DL
31	4-21-05	ISSUED AS-BUILT PER PC/M 04-056.	RH	JPG	AAP	CMZ	37	9/28/09	ISSUED AS-BUILT PER CRN-I-5092 (PC/M 09-070).	RV	RH	–	LG
42	4-7-15	ISSUED AS-BUILT PER EC 283531.	RH	BB	–	SB	36	9-22-09	ISSUED AS-BUILT PER CRN M-12837 (PC/M 09-070).	RH	RV	–	LG
41	4-24-13	ISSUED AS-BUILT PER EC 246978 (ITOP 13-04-164).	RH	BB	–	JM	35	9/25/06	ISSUED AS-BUILT PER CRN-M-11759 (PC/M 03-106).	RH	RV	BSG	JTL
40	3-4-13	ISSUED AS-BUILT PER EC 246924 (ITOP 13-03-010).	RV	RH	–	JD	34	7-11-05	ISSUED AS-BUILT PER CRN-M-11395 (PC/M 03-106).	RH	RRW	AAP	PRB
39	2-10-13	ISSUED AS-BUILT PER EC 246924 (ITOP'S 13-02-043 & 047). (PARTIAL)	RV	BB	PSB	CW	33	6-28-05	ISSUED AS-BUILT PER CRN M-11337 (PC/M 04-065).	RH	RV	RSV	ASD

SAFETY RELATED

POD

TURKEY POINT NUCLEAR UNIT 4

P & ID

REACTOR COOLANT SYSTEM

STONE & WEBSTER ENGINEERING CORP. FT. LAUDERDALE, FLORIDA

FPL

DRAWING NUMBER	SYS
5614-M-3041	041
SHEET 2	REV. 42

Question # 35

Given the following:

- Unit 4 is at 100% power

Subsequently:

- 4B CCW Heat Exchanger develops a tube leak.

Which one of the following completes the statements below?

A __(1)__ CCW Head Tank Level will confirm 4B CCW tube leak.

Upon isolation of the 4B CCW Heat Exchanger, Unit 4 __(2)__required to enter an action statement for Tech Spec 3.7.2, Component Cooling Water.

A. (1) rising
(2) is

B. (1) rising
(2) is NOT

C. (1) lowering
(2) is

D. (1) lowering
(2) is NOT

Answer Analysis

A. Incorrect.
   Part 1 incorrect, plausible since other HX tube failures will cause the CCW Head Tank to rise, i.e. NRHX tube leak.
   Part 2 incorrect, plausible since OPERABLE CCW HXs are required for TS 3.7.3.

B. Incorrect.
   Part 1 incorrect, plausible since other HX tube failures will cause the CCW Head
   Part 2 correct.

C. Incorrect.
   Part 1 correct.
   Part 2 incorrect, plausible since OPERABLE CCW HXs are required for TS 3.7.3.

D. Correct. CCW Head Tank level will lower due to CCW system having a higher pressure than ICW system. TS 3.7.3 only requires 2 CCW HX OPERABLE, hence not TS action statement entry required.

Question Number: 35

Tier: 2 **Group:** 1

K/A: 008A4.10

Ability to manually operate and/ or monitor, in the control room, conditions that require the operation of two CCW coolers: Note: LOCAs are conditions that require operation of two CCW Hxs as each is rated at 50%

Importance Rating: **3.1**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:**

K/A Match: **RO question - TS apects are above the line knowledge.** The K/A is the ability to operate / monitor CCW for conditions that require two operating CCW coolers. A LOCA requires operation of each of the two 50% CCW coolers. Per Technical Specifications 3 CCW heat exchangers must be operable to ensure capability in the event of a Single failure and a LOCA condition. It is matched as the applicant must evaluate plant conditions, identify operator action per 0NOP-030 required to mitigate the event and recognize a 1-hour TS action. UFSAR section 9 the CCW System.

SRO Justification: N/A

Technical References: ARP H7/6, Technical Specification 3.7.2
Lesson Plan 6902140, Component Cooling Water

Proposed references to be provided: NONE

Learning Objective: PTN 6902140 OBJ. 6 & 8

Cognitive Level: Higher X

Lower _

Question Source: New

Modified Bank X_

Bank _

Question History: Turkey Point 2017

Comments: Screenshot of development references as applicable

3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps, and

b. Two CCW heat exchangers.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With less than two CCW heat exchangers OPERABLE, restore two heat exchangers to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.

CAUSES:
1. Leak out of CCW System
2. Seal Water Heat Exchanger tube leak.

H7/6

CCW SURGE TANK LO LEVEL

DEVICE: LT-4-613

SETPOINT: Low - 45%

LOCATION: CCW Surge Tank in SFP Room

ALARM CONFIRMATION

CHECK CCW Surge Tank level indication, LI-4-613A on VPB.

OPERATOR ACTIONS

1. IF low level, THEN **OPEN** Fill Valve, MOV-4-832 to restore level to normal (greater than 45% in CCW Surge Tank when CCW System is drained for maintenance).
2. IF CCW Surge Tank level can **NOT** be maintained, AND in MODES 1 through 4, THEN:
   A. **REFER TO** 4-ONOP-030, Component Cooling Water Malfunction.
   B. **REVIEW** Tech Spec 3.0.3 for applicability.
   C. **REFER TO** AR 2061032 for determining CCW System operability.
3. **CHECK** CCW and supplemental CCW Systems for possible leaks.

NOTE

A low level concurrent with any of the following may indicate a Seal Water Heat Exchanger Tube leak:
- Annunciator A-4/6 (VCT HI/LO LEVEL)
- Increased VCT level
- Unexplained slight power\Tave increase
- Increased sodium levels in the RCS

4. **ISOLATE** source of leakage.

REFERENCES:
1. FPL DWG 5614-M-3030
2. PC/M 96-093, Addition of U-4 CCW Head Tank
3. Tech Spec 3.7.2

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	026		2.2.42
	Importance Rating	3.9		

Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question: RO Question # 6

Given the following conditions:

- Unit 3 is at 100% power.

Subsequently the following events take place:

- 3B CCW Heat Exchanger develops a tube leak and is removed from service.
- 3C CCW Pump fails to manually start when demanded by the RO.

Which one of the following identifies the technical specification (TS) implications?

A TS action statement entry __(1)__ required for the 3B CCW Heat Exchanger.

A TS action statement entry __(2)__ required for the 3C CCW Pump.

A.	(1) is (2) is
B.	(1) is (2) is NOT
C.	(1) is NOT (2) is
D.	(1) is NOT (2) is NOT

Proposed Answer: C

A.	**Incorrect** Part 1 is incorrect but plausible since some LCOs require all redundant components to be operable in MODE 1 (e.g. all RHR HX exchangers required to be OPERABLE)
B.	**Incorrect** Both parts incorrect but plausible per discussion below. Question parts are independent and plausible as a whole.
C.	**Correct** MODE 1 CCW system LCO requires 2 CCW HXs and 3 CCW pumps to be OPERABLE. In this case 2 CCW HXs and 2 CCW pumps remain operable.
D.	**Incorrect** Part 2 is incorrect but plausible since some LCOs require only one redundant component to be operable in MODE 1 (e.g. 2 of 3 charging pumps required)

Technical Reference(s)	TS 3.7.2		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902140 obj 8		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		8
	55.43		
Components, capacity, and functions of emergency systems.			

Question # 36

Given the following:

- Unit 3 is performing a plant heat up post Refueling Outage.
- RCS Tave is 300°F and stable.
- PRZ pressure is 1000 psig and stable.

Subsequently:

- One PZR PORV starts leaking.
- PRT pressure is 10 psig

Which one of the following completes the statement below?

The expected PORV tail pipe temperature is _______.

A. 193°F

B. 240°F

C. 300°F

D. 545°F

Answer Analysis

A. Incorrect. Plausible, since this is the saturation temperature for 10 psia.

B. Incorrect. Plausible since this is the typical value for a isenthalpic process during a leaking PORV from PZR NOP/NOT. If candidate remembers only the value and doesn't apply the process, he will not come to the correct answer.

C. Correct. Expansion from 1000 psia to 10 psig in the tailpipe will result in a tailpipe temperature of approximately 300°F due to isenthalpic expansion through the Safety Valve. By providing that the annunciator is in alarm, the candidate will have to understand that the PRT pressure is approximately 10 psig.

D. Incorrect. Plausible, since this is the saturation temperature for 1000 psig.

Question Number: 36

Tier: 2 **Group:** 1

K/A: 010 (SF3 PZR PCS) Pressurizer Pressure Control

010K5.02; Knowledge of the operational implications of the following concepts as

the apply to the PZR PCS:

Constant enthalpy expansion through a valve

Importance Rating: **2.6**

10 CFR Part 55: 41.5

10 CFR 55.43.b **:** XXX

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: Steam Tables

Proposed references to be provided: None

Learning Objective: PTN 6902109 Objective 11

Cognitive Level: Higher _

Lower _

Question Source: New _

Modified Bank X

Bank _

Question History: This question was derived from the 2017 PTN exam, RO question #36. Significant modification made to plant conditions to change answer choices and final answer.

Comments: Screenshot of development references as applicable

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	010		K5.02
	Importance Rating	2.6		

Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Constant enthalpy expansion through a valve

Proposed Question: RO Question # 36

Given the following conditions:

- Unit 3 is in MODE 3.
- RCS Tave is 540°F and stable.
- PRZ pressure is 1800 psig and stable.

Subsequently:

- A PZR PORV starts leaking.
- ANN A7/1, PRT HI/LO LEVEL HI PRESS/TEMP, alarms.
- PI-3-472, PRT Pressure, indicates 18 psig and rising.

Which one of the following completes the statement below?

The nearest expected PORV tail pipe temperature is ______.

A.	222°F
B.	256°F
C.	621°F
D.	653°F

Proposed Answer: B

A.	**Incorrect** Plausible, since this is the saturation temperature for 18 psia.
B.	**Correct** Expansion from 1800 psia to 18 psig in the tailpipe will result in a tailpipe temperature of approximately 255°F due to isenthalpic expansion through the Safety Valve.
C.	**Incorrect** Plausible, since this is the saturation temperature for 1800 psig.
D.	**Incorrect** Plausible, since this is the saturation temperature for the normal operating pressure.

Technical Reference(s)	LP 6902109		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Steam Tables
Learning Objective:	6902109 Objective 11		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		14
	55.43		
Principles of heat transfer, thermodynamics and fluid mechanics.			
Comments:			

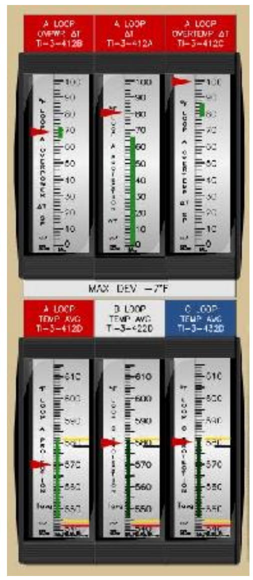
Question # 37

Given the following:

- Unit 3 is at 100% power.

Subsequently:

- ANN J7/5, RTD CHANNEL I FAILURE, alarms.
- ANN J7/4, EAGLE 21 TROUBLE, alarms.
- The RO observes the following indication on VPA.



ALSO PROVIDED AS A REFERENCE IN LARGER FORMAT

Which one of the following completes the statements below?

Assuming NO operator actions:

- The RPS OPDT is __(1)__ bi-stable(s) away from generating a reactor trip.
- The RPS OTDT is __(2)__ bi-stable(s) away from generating a reactor trip.

A. (1) 1
(2) 1

B. (1) 1
(2) 2

C. (1) 2
(2) 1

D. (1) 2
(2) 2

Answer Analysis

A. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since the OTDT trip setpoint is outside the green band.

B. Correct. Since the Delta T indication has risen, it is above the actuation setpoint for OPDT. With the lower Tave the actuation setpoint for OTDT has a larger margin now and the logic for actuation remains 2/3.

C. Incorrect.
Part 1 incorrect, plausible since the OPDT indication remains in the green band.
Part 2 incorrect, plausible since the OTDT trip setpoint is outside the green band.

D. Incorrect.
Part 1 incorrect, plausible since the OPDT indication remains in the green band.
Part 2 correct.

Question Number: 37

Tier: 2 **Group:** 1

K/A: 012K6.06

Knowledge of the effect or loss of Sensors and detectors on the Reactor Protection (RPS) system

Importance Rating: **2.7**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, identify the calculated OTΔT system response and operator action required to mitigate the event.

SRO Justification: N/A

Technical References: Lesson Plan PTN 6902163, RPS and Safeguards Actuation systems

Proposed references to be provided: None

Learning Objective: PTN 6902136 OBJ. 6

Cognitive Level: Higher X

Lower _

Question Source: New X

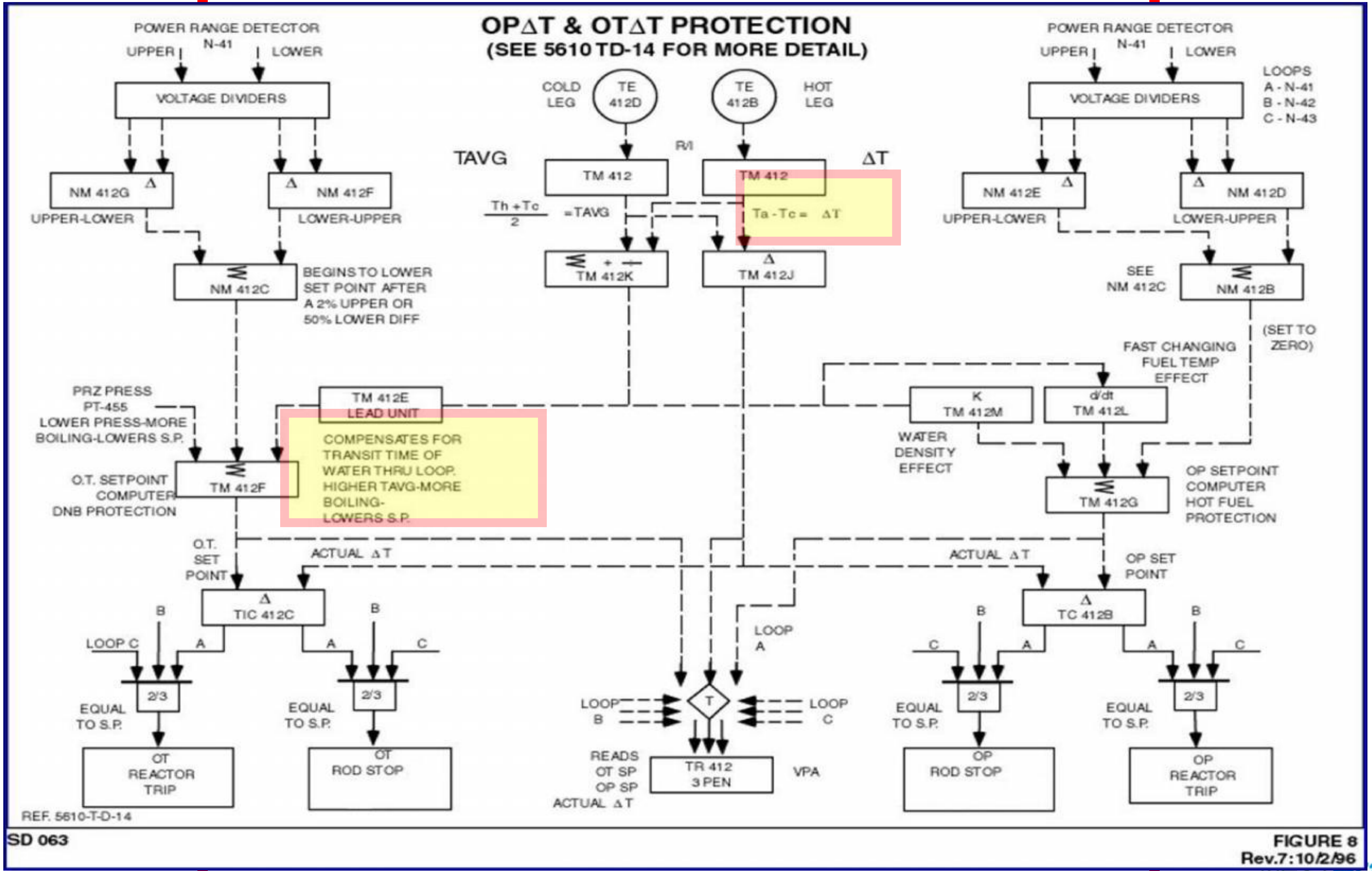
Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

Overpower and Overtemperature ΔT Trips (Fig 8)



Objectives 6,7

Question # 38

Which one of the following completes the statements below?

A Reactor Trip signal __(1)___ AUTOMATICALLY place the Control Room on recirc. A Reactor Trip signal __(2)___ trip the MG sets.

A. (1) will
   (2) will

B. (1) will
   (2) will NOT

C. (1) will NOT
   (2) will

D. (1) will NOT
   (2) will NOT

Answer Analysis

A. Incorrect.
   Part 1 incorrect, plausible since a reactor trip coincident with a safety injection will place the control room on recirc.
   Part 2 incorrect, plausible since tripping of the MG Sets will cause same response on the plant as a reactor trip.

B. Incorrect.
   Part 1 incorrect, plausible since a reactor trip coincident with a safety injection will place the control room on recirc.
   Part 2 correct.

C. Incorrect.
   Part 1 correct.
   Part 2 incorrect, plausible since tripping of the MG Sets will cause same response on the plant as a reactor trip.

D. Correct. The control room is not placed on recirc from ONLY a reactor trip. MG sets do not receive a trip signal from the RPS system.

Question Number: 38

Tier: **2** **Group:** 1

K/A: 012 (SF7 RPS) Reactor Protection

012G2.1.27; Knowledge of system purpose and/or function.

Importance Rating: **3.9**

10 CFR Part 55: 41.7

10 CFR 55.43.b : XXX

K/A Match:

SRO Justification: N/A

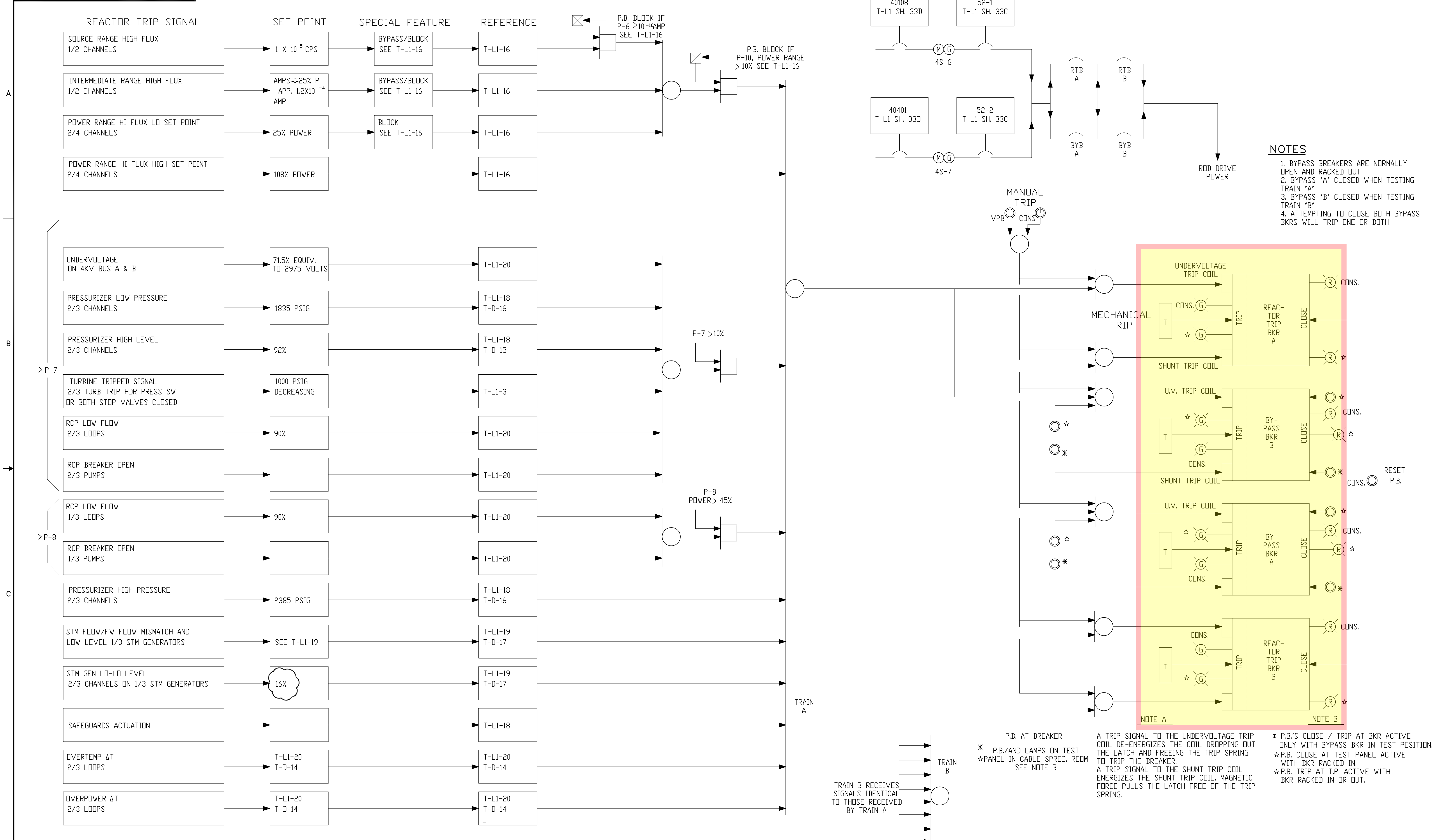
Technical References: 5610-T-L1 SH-2

Proposed references to be provided: None

Learning Objective: PTN 6902163 OBJ. 6 & 7

Cognitive Level: Higher _

	Lower	X
Question Source:	New	X
	Modified Bank	_
	Bank	_
Question History:	New	
Comments:	Screenshot of development references as applicable	



NOTES

1. BYPASS BREAKERS ARE NORMALLY OPEN AND RACKED OUT
2. BYPASS 'A' CLOSED WHEN TESTING TRAIN 'A'
3. BYPASS 'B' CLOSED WHEN TESTING TRAIN 'B'
4. ATTEMPTING TO CLOSE BOTH BYPASS BKRS WILL TRIP ONE OR BOTH

P.B. AT BREAKER

* P.B./AND LAMPS ON TEST
☆ PANEL IN CABLE SPRED. ROOM SEE NOTE B

A TRIP SIGNAL TO THE UNDERVOLTAGE TRIP COIL DE-ENERGIZES THE COIL DROPPING OUT THE LATCH AND FREEING THE TRIP SPRING TO TRIP THE BREAKER.
A TRIP SIGNAL TO THE SHUNT TRIP COIL ENERGIZES THE SHUNT TRIP COIL. MAGNETIC FORCE PULLS THE LATCH FREE OF THE TRIP SPRING.

* P.B.'S CLOSE / TRIP AT BKR ACTIVE ONLY WITH BYPASS BKR IN TEST POSITION.
☆ P.B. CLOSE AT TEST PANEL ACTIVE WITH BKR RACKED IN.
☆ P.B. TRIP AT T.P. ACTIVE WITH BKR RACKED IN OR OUT.

REV	DATE	REVISION	BY	CH	APP	APP
1	07-16-13	ISSUED AS-BUILT PER CRN-015 (EC 247049) (ITOP 13-07-037).	RV	RH	–	PMB
0	3/19/13	THIS DRAWING SUPERSEDES DWG. 5610-T-L1 SH 2. ISSUED AS-BUILT PER EC 246859, INCORP. CRN-171 AND ENTERED INTO FPL SYSTEM. (ITOP 13-03-178)	RV	BB	–	PMB

SAFETY RELATED

POD

TURKEY POINT NUCLEAR UNIT 4

LOGIC DIAGRAM

REACTOR TRIP SIGNALS

FLORIDA POWER & LIGHT

DRAWING NUMBER 5614-T-L1

SYS –

SHEET: 2

REV 1

5614-T-L1_SH_2.DWG

Question # 39

Which one of the following completes the statements below?

The 3B 120 VAC Vital Inverter is NORMALLY aligned to the __(1)__.

The 3C 120 VAC Vital Inverter is NORMALLY aligned to the __(2)__.

A. (1) 3D01 Vital DC Bus
(2) 3D23 Vital DC Bus

B. (1) 3D01 Vital DC Bus
(2) 4D23 Vital DC Bus

C. (1) 4D01 Vital DC Bus
(2) 3D23 Vital DC Bus

D. (1) 4D01 Vital DC Bus
(2) 4D23 Vital DC Bus

Answer Analysis

Discussion: Table below shows the scheme. This is knowledge of power supplies question / KA. 3D01 powers A inverter, 3D23 powers C inverter, and 4D01 powers B inverter and 4D23 powers D inverter through 3/4 P07, 06, 08, 09 buses respectively. See table below

A. Incorrect, see discussion above

B. Incorrect, see discussion above

C. Correct, see discussion above

D. Incorrect, see discussion above

Question Number: 39

Tier: 2 **Group:** 1

K/A: **013K2.01**

Knowledge of bus power supplies to ESFAS / safeguards equipment control

Importance Rating: **3.6**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must demonstrate knowledge of the power supplies to Safeguards actuation logic and instrumentation power.

SRO Justification: N/A

Technical References: 5610-T-E-1592 SH-2

Proposed references to be provided: None

Learning Objective: PTN 6902139 OBJ. 4

Cognitive Level: Higher __

Lower X_

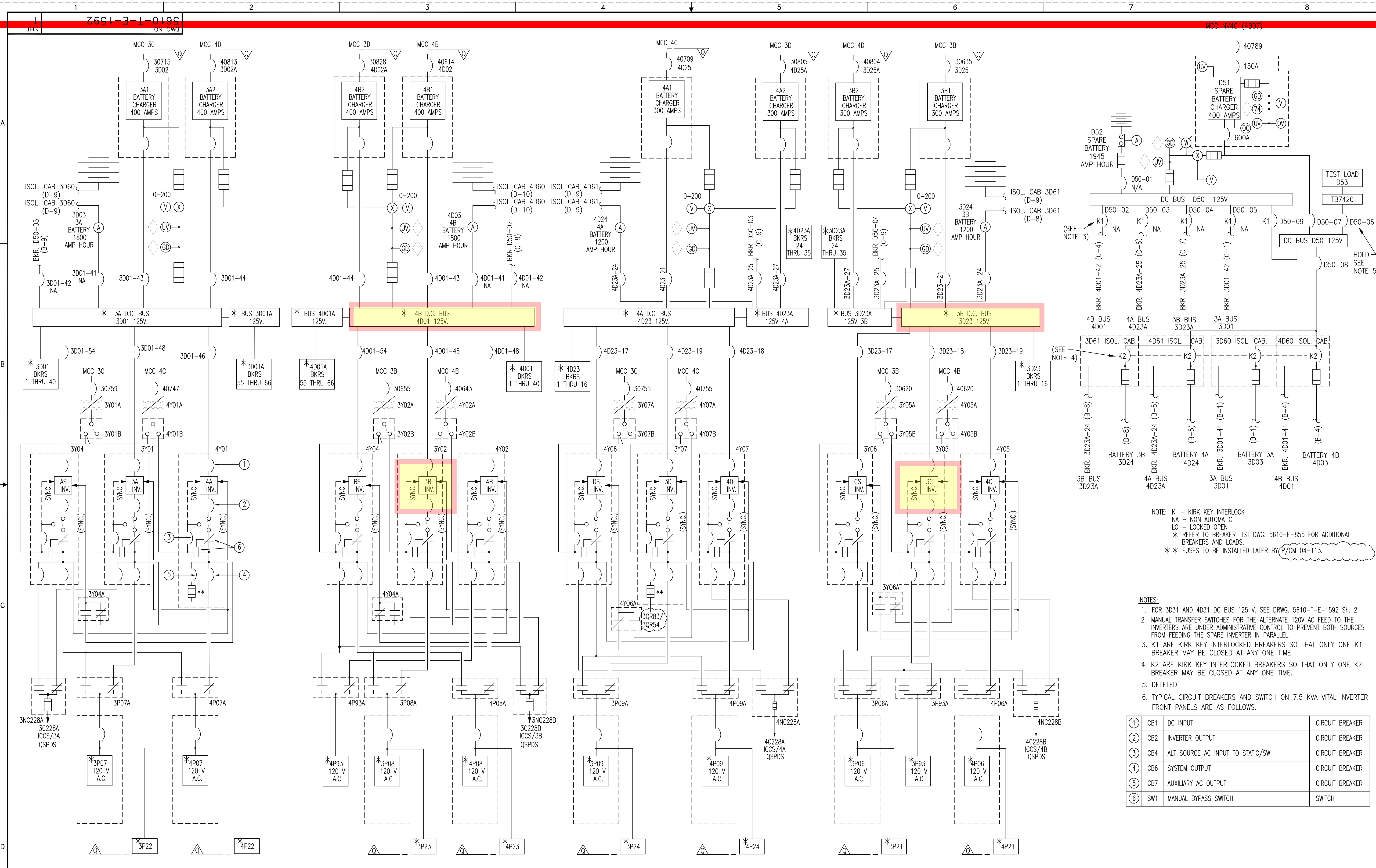
Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable



NOTES:

1. FOR 3D31 AND 4D31 DC BUS 125 V. SEE DRWG. 5610-T-E-1592 Sh. 2.
2. MANUAL TRANSFER SWITCHES FOR THE ALTERNATE 120V AC FEED TO THE INVERTERS ARE UNDER ADMINISTRATIVE CONTROL TO PREVENT BOTH SOURCES FROM FEEDING THE SPARE INVERTER IN PARALLEL.
3. K1 ARE KIRK KEY INTERLOCKED BREAKERS SO THAT ONLY ONE K1 BREAKER MAY BE CLOSED AT ANY ONE TIME.
4. K2 ARE KIRK KEY INTERLOCKED BREAKERS SO THAT ONLY ONE K2 BREAKER MAY BE CLOSED AT ANY ONE TIME.
5. DELETED
6. TYPICAL CIRCUIT BREAKERS AND SWITCH ON 7.5 KVA VITAL INVERTER FRONT PANELS ARE AS FOLLOWS.

1	CB1	DC INPUT	CIRCUIT BREAKER
2	CB2	INVERTER OUTPUT	CIRCUIT BREAKER
3	CB4	ALT SOURCE AC INPUT TO STATIC/SW	CIRCUIT BREAKER
4	CB6	SYSTEM OUTPUT	CIRCUIT BREAKER
5	CB7	AUXILIARY AC OUTPUT	CIRCUIT BREAKER
6	SW1	MANUAL BYPASS SWITCH	SWITCH

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
45	12-18-12	ISSUED AS-BUILT PER EC 242033. (ITOP 12-12-020)	RV	BB	–	BAP	40	07-13-04	ISSUED AS-BUILT PER CRN-E-16624 (PC/M 03-106).	RH	RV	FJK	TS
44	12/22/08	ISSUED AS-BUILT PER PC/M 04-048.	RV	BB	–	PMB	39	07-31-01	ISSUED AS-BUILT PER CRN-E-16236 (PC/M 00-016).	RV	RH	MCP	TAB
43	11/24/08	ISSUED AS-BUILT PER PC/M 04-048. (PARTIAL)	RV	DAR	–	PMB	38	10-21-97	ISSUED AS-BUILT PER CRN'S E-15712 & E-15708(PC/M 97-028).	RV	RH	FJK	RB
42	7/25/07	ISSUED AS-BUILT PER PC/M 03-109.	RV	RH	–	PDS	37	4-25-94	INCORP. CRN-E-13300 FOR PC/M 93-047.	RV	RH	BAW	JCW
41	5/3/06	ISSUED AS-BUILT PER PC/M 03-135.	RV	RH	FJK	PMB	36	7/29/91	ISSUED AS-BUILT FOR PC/M's 83-06 & 87-265.	RH	LH	–	JG

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

OPERATING DIAGRAM

125V D.C. & 120V INSTRUMENT A.C. ELECTRICAL DISTRIBUTION

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-E-1592

SYS –

REV 45

SHEET: 1

5610-T-E-1592_SH 1.DWG

Question # 40

Given the following:

- The RO MANUALLY trips the reactor.

Subsequently:

- An inadvertent Containment Isolation phase B signal is initiated.

Which one of the following completes the statements below?

The RCPs are __(1)__ due to the __(2)__.

A. (1) expected to AUTOMATICALLY trip
(2) initiation of containment spray

B. (1) expected to AUTOMATICALLY trip
(2) loss of CCW to the RCPs

C. (1) required to be MANUALLY tripped IAW 3-EOP-E-0, Reactor Trip or Safety Injection
(2) initiation of containment spray

D. (1) required to be MANUALLY tripped IAW 3-EOP-E-0, Reactor Trip or Safety Injection
(2) loss of CCW to the RCPs

Answer Analysis

A. (1) Incorrect, Plausible if the candidate thinks the containment isolation signal will cause an automatic trip the RCPs
(2) Incorrect, Plausible if the candidate believes that RCP's must be stopped due to containment spray actuation.

B. (1) Incorrect, see A(1)
(2) Correct, Per procedure requirements of the EOP-E-0 foldout and design basis explain that RCP's are to be tripped when cooling is lost to the RCP seals and motors to prevent overheating.

C. (1) Correct, there is no automatic protective trip associated with loss of CCW or CIS signal
(2) Incorrect, see B(2)

D. Correct, see discussion above

Question Number: 40

Tier: 2 **Group:** 1

K/A: 013 (SF2 ESFAS) Engineered Safety Features Actuation

013K3.02; Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following:

RCS

Importance Rating: **4.3**

10 CFR Part 55: 41.7

10 CFR 55.43.b : XXX

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-EOP-E-0, BD-EOP-E-0

Proposed references to be provided: None

Learning Objective: PTN 6902321 OBJ. 4

Cognitive Level: Higher X _

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

BASIS DOCUMENT

PWROG Guideline Step 14 **PTN Procedure Step** ATT. 3 – Step 10

Verify Containment Spray NOT Required

BASIS:

If containment pressure exceeds the high-high setpoint, containment spray is automatically initiated to mitigate the containment pressure transient. Containment isolation phase B valves are closed to isolate additional potential release paths from containment. Since component cooling to the RCP seals and motors is isolated on a phase B signal, the RCPs are tripped to preclude overheating of the seals and motors.

The basis for monitoring containment pressure is to ensure that the operator verifies system operation any time the containment spray setpoint is exceeded thus taking necessary action to maintain containment integrity and prevent damage to the RCPs.

STEP DEVIATIONS FROM PWROG GUIDELINES:
TYPE DESCRIPTION

9 A reference to specific plant instrumentation was added to ensure the operator looks at a history of containment pressure. This was done in response to an EOP validation comment.

7 The substep to verify containment isolation phase B valves closed was modified to use installed status lights designed for this purpose.

9 The IF, NOT, THEN logic statements were rewritten to comply with plant specific writers guide requirements.

9 The RNO was modified to provide guidance for isolation of any containment penetration which does not automatically isolate.

PLANT SPECIFIC SETPOINTS:

20 psig Containment pressure setpoint for spray actuation. (EOP Setpoint T.2)

FOLDOUT PAGE
For Procedure 3-EOP-E-0

1. **ADVERSE CONTAINMENT CONDITIONS**
   A. IF either condition listed below occurs, THEN use [Adverse Containment Setpoints]:
      * Containment atmosphere temperature ≥ 180°F

        OR

      * Containment radiation levels ≥ $1.3x10^5$ R/hr

   B. WHEN Containment atmosphere temperature returns to less than 180°F, THEN Normal Setpoints can again be used.

   C. WHEN Containment radiation levels return to less than $1.3x10^5$ R/hr, THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.

2. **RCP TRIP CRITERIA**
   - IF both conditions listed below occur, THEN trip all RCPs:
     1) High-Head SI Pumps – AT LEAST ONE RUNNING AND SI FLOWPATH VERIFIED.
     2) RCS subcooling – LESS THAN 19°F[41°F].
   - IF Phase B actuated, THEN trip all RCPs.

3. **FAULTED S/G ISOLATION CRITERIA**

   IF any S/G pressure decreasing in an uncontrolled manner OR any S/G completely depressurized, THEN the following shall be performed:

   A. Maintain total feedwater flow greater than 400 gpm until narrow range level in at least one S/G is greater than 7%[27%].

   B. IF any S/G(s) are **NOT** faulted, THEN Isolate AFW flow to faulted S/G(s).
      1) IF **NOT** able to isolate AFW flow to any faulted S/G(s), THEN locally isolate AFW using Attachment 6.
      2) Stabilize RCS Hot Leg temperature using Steam Dumps when faulted S/G has blown down to less than 9% Wide Range [27% Narrow Range] by reducing intact S/G Steam Dump To Atmosphere valves AUTO setpoint to match current S/G pressure.

4. **RUPTURED S/G ISOLATION CRITERIA**

   IF any S/G level increases in an uncontrolled manner OR any S/G has abnormal radiation, AND Narrow Range Level in affected S/G(s) is greater than 7%[27%], THEN feed flow may be stopped to affected S/G(s).

5. **AFW SYSTEM OPERATION CRITERIA**

   A. IF two AFW Pumps are operating on a single train, THEN one of the pumps shall be shut down within one hour of the initial start signal.

   B. IF two AFW Trains are operating AND one of the AFW Pumps has been operating at low flow of 80 gpm or less for one hour, THEN that AFW Pump shall be shut down.

6. **CST MAKEUP WATER CRITERIA**

   IF CST level decreases to less than 12%, THEN add makeup to CST using 3-NOP-018.01, CONDENSATE STORAGE TANK (CST).

7. **RHR SYSTEM OPERATION CRITERIA**

   IF RHR flow is less than 1100 gpm, THEN the RHR Pumps shall be shut down within 44 minutes of the initial start signal.

8. **LOSS OF OFFSITE POWER OR SI ON OTHER UNIT**

   IF SI has been reset AND subsequently either offsite power is lost OR SI actuates on the other unit, THEN restore safeguards equipment and at least one Computer Room Chiller to required configuration using Attachment 3.

   Refer to Attachment 2 for essential loads.

9. **LOSS OF CHARGING CRITERIA**

   IF charging capability has been lost, AND high-head SI Pumps are running at shutoff head, THEN rotate High-Head SI Pumps as necessary to maintain continuous run time of any pump less than 30 minutes while maintaining at least one High-Head SI Pump running.

Question # 41

Given the following:

- Unit 4 experiences an AUTOMATIC Safety Injection.
- All RCPs are tripped IAW the foldout page of 4-EOP-E-0, Reactor Trip or Safety Injection.
- The crew is performing Attachment 3 of 4-EOP-E-0.

Which one of the following completes the statements below?

The 4A ECC __(1)__ AUTOMATICALLY start.

IAW Attachment 3 of 4-EOP-E-0, the RO __(2)__ MANUALLY start the Normal Containment Coolers.

A. (1) will
   (2) will

B. (1) will
   (2) will NOT

C. (1) will NOT
   (2) will

D. (1) will NOT
   (2) will NOT

Answer Analysis

Discussion: This is a modified TP 2015 exam question which was on Unit 3 and had a bus failure. This question is on Unit 4 and does NOT have a bus failure. Correct Answer is D.

On a reactor trip and SI due to a LOCA in containment the NCC coolers do not auto start and must be manually reset and restarted. However per E-0, Attachment 3, if the RCPs are secured then manually restarting the NCCs is bypassed at step 3 of the attachment to avoid overheating CCW. <u>On Unit 4 ONLY the B & C ECCs auto start unless they have lost power or CCW is isolated.</u> On Unit 3 the A and C ECCs auto start. Attachment 3 requires verification that ONLY two (2) ECCs are running.

A. Incorrect.
Part 1 incorrect, plausible since on Unit 3, the 3A ECC Normally receives a start signal.
Part 2 incorrect, plausible since Plausible because it is correct if RCPs are running.

B. Incorrect.
Part 1 incorrect, plausible since on Unit 3, the 3A ECC Normally receives a start signal.
Part 2 incorrect, plausible since Plausible because it is correct if RCPs are running.

C. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since Plausible because it is correct if RCPs are running.

D Correct; See Discussion above

Question Number: 41

Tier: 2 **Group:** 1

K/A: **022A4.01 Ability to manually operate and/or monitor** CCS fans **in the control room**
Note the KA reference to CCS fans could be either or both NCC and ECC containment fans. Question addressed both by asking how each would be operated under the conditions presented

Importance Rating: **3.6**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, identify the RCPs are tripped, automatic system response and be aware that NCCs are not operated when RCPs are tripped to conserve CCW loads (purpose and overall mitigation strategy)

SRO Justification: N/A

Technical References: E-0, Attachment 3, 5614-T-L1 SH 12A/B
Final Safety Analysis Report, Sections 5.1, 6.3, 9.8, 9.10, 14
Lesson Plan 6902129, Containment Vent and Heat Removal System

Proposed references to be provided: None

Learning Objective: PTN 6902321 OBJ. 4, PTN 6902332 Obj. 2, 3

Cognitive Level: Higher X

Lower _

Question Source: _

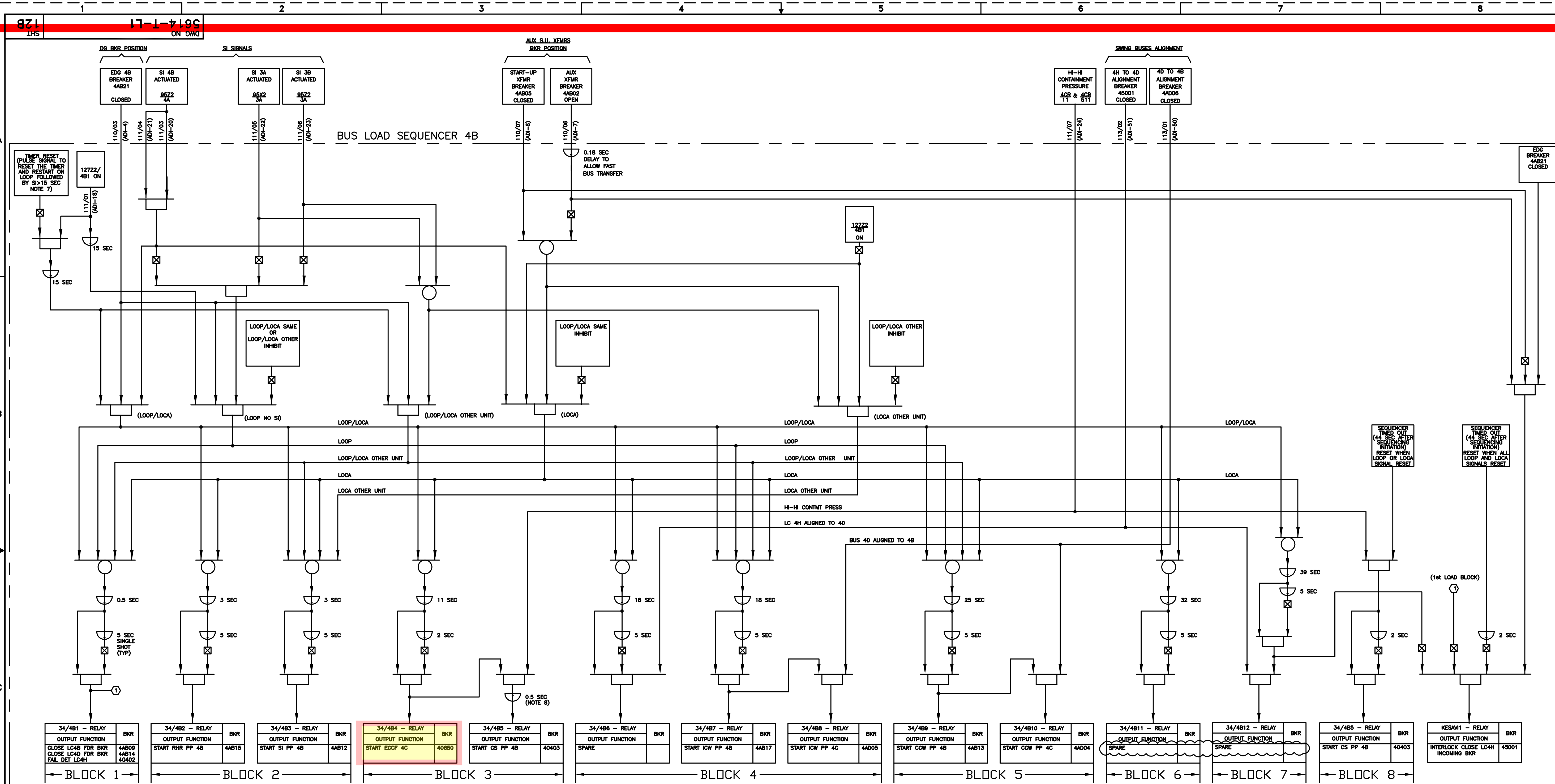
Modified Bank X_
Bank _

Question History: Turkey Point 2011

Comments: Screenshot of development references as applicable

ATTACHMENT 3
Prompt Action Verifications
(Page 11 of 13)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16.	**Reestablish RCP Cooling**	
	a. Check RCPs – AT LEAST ONE RUNNING	**a.** Go to Attachment 3, Step 17.
	b. Open CCW To Normal Containment Cooler Valves: • MOV-4-1417 • MOV-4-1418	**b.** Stop all RCPs.
	c. Reset and start Normal Containment Coolers	**c.** Stop all RCPs.
17.	**Verify Control Room Ventilation Isolation**	
	a. Verify Emergency Air Supply Fans – AT LEAST ONE RUNNING: * SF-1A * SF-1B	**a.** Manually start one Emergency Air Supply Fan.
	b. Control Room Ventilation dampers – ALIGNED FOR RECIRC	**b.** Manually align dampers for Recirculation.
	c. Verify Normal Flow green indicating light (4QR82) – ON	**c.** Manually start a second Emergency Air Supply Fan.
	d. TS-0002, TSC Emergency Vent Auto Initiate Key Switch – IN ENABLE	**d.** Place switch in ENABLE.
18.	**Place Hydrogen Monitors In Service Using 4-NOP-094, CONTAINMENT POST ACCIDENT MONITORING SYSTEM**	



BUS LOAD SEQUENCER 4B

***	LOAD CONDITIONS (EVENT)	1st LOAD BLOCK	2nd LOAD BLOCK		3rd LOAD BLOCK		4th LOAD BLOCK			5th LOAD BLOCK		6th LOAD BLOCK	7th LOAD BLOCK	8th LOAD BLOCK	LC 4H TRANSFER PERMISSIVE
1	LOOP	X						X	X	X	X				THE KESAVI1 RELAY IS NORMALLY DE-ENERGIZED ALLOWING FOR LC-4H TRANSFER. THE RELAY ENERGIZES AS SOON AS THE AUX AND S/U XFMR BKRS ARE OPEN AND THE EDG 4B BREAKER CLOSES. IT THEN DE-ENERGIZES DURING THE FIRST LOAD BLOCK, ENERGIZES AFTER THE FIRST LOAD BLOCK, DE-ENERGIZES DURING THE SEVENTH LOAD BLOCK, ENERGIZES AFTER THE SEVENTH LOAD BLOCK, AND DE-ENERGIZES TWO SECONDS AFTER THE SEQUENCER HAS TIMED OUT.
2	LOOP WITH A SIMULTANEOUS LOCA SAME TRAIN (4B)	X	X	X	X		X	X	X	X	X	X	X		
3	LOOP FOLLOWED SOMETIME LATER BY A LOCA SAME TRAIN (4B)	X	X	X	X		X	X	X	X	X	X	X		
4	LOOP WITH A SIMULTANEOUS LOCA OTHER UNIT	X		X				X	X	X	X				
5	LOOP FOLLOWED SOMETIME LATER BY A LOCA OTHER UNIT	X		X				X	X	X	X				
6	LOCA SAME TRAIN (4B)	X	X	X	X		X	X	X	X	X	X	X		
7	LOCA OTHER UNIT			X											
8	HI-HI CONTAINMENT PRESSURE CONCURRENT OR LESS THAN 13 SECONDS AFTER A LOOP/LOCA	X	X	X	X	X	X	X	X	X	X	X	X		
9	HI-HI CONTAINMENT PRESSURE CONCURRENT OR LESS THAN 13 SECONDS AFTER A LOCA	X	X	X	X	X	X	X	X	X	X	X	X		
10	HI-HI CONTAINMENT PRESSURE OCCURING MORE THAN 13 SECONDS AFTER A LOCA	X	X	X	X		X	X	X	X	X	X	X	X	
11	HI-HI CONTAINMENT PRESSURE OCCURING MORE THAN 13 SECONDS AFTER A LOOP/LOCA	X	X	X	X		X	X	X	X	X	X	X	X	
12	LOCA SAME TRAIN FOLLOWED SOMETIME LATER BY LOOP	X	X	X	X		X	X	X	X	X	X	X		
13	LOCA OTHER UNIT FOLLOWED SOMETIME LATER BY LOOP	X		X				X	X	X	X				

NOTES:

1. THIS DRAWING REPRESENTS A CONCEPTUAL LOGIC DIAGRAM OF THE SEQUENCING LOADING FUNCTION OF THE EMERGENCY BUS LOAD SEQUENCER 4B.
2. DELETED
3. DELETED
4. THE EIGHTH LOAD BLOCK REPRESENTS A LATE START OF THE CONTAINMENT SPRAY PUMP DUE TO A LATE RECEIPT (TIME LATER THAN 13 SECONDS AFTER LOCA OR LOOP/LOCA) OF A HIGH-HIGH CONTAINMENT PRESSURE SIGNAL.
5. DELETED.
6. THE BLOCKS SHOWN INSIDE THE SEQUENCER BOX ARE GENERATED INTERNALLY BY THE SEQUENCER LOGIC.
7. ON LOOP FOLLOWED BY LOCA SCENARIO, THE TIMER IS RESET AND RESTARTS, IF SI OCCURS IN MORE THAN 15 SEC'S AFTER LOOP.
8. TIME DELAY PROVIDED TO ENSURE CONTAINMENT SPRAY PUMP START IF HI-HI CONTAINMENT PRESSURE IS RECEIVED NEAR END OF TWO SECOND LOAD BLOCK 3 PULSE.

NOTE: THIS DWG IS MADE FROM:
__________(VENDOR) DWG. NO.__________
__________(A/E) DWG. NO.__________REV.____
FPL DWG NO. 5610-T-L1, SH.12B REV.15.

SAFETY RELATED | POD

REV	DATE	REVISION	BY	CH	APP	APP
3	2-16-13	ISSUED AS-BUILT PER EC 246931 (ITOP 13-02-097).	RV	BB	CJF	RP
2	10-21-96	ISSUED AS-BUILT PER PC/M 95-148.	RV	RH	BP	TS
1	3-25-96	ISSUED AS-BUILT FOR PC/M 95-033.	RV	RH	AD	SGB
0	9-22-95	ISSUED AS-BUILT PER PC/M 95-032 AND ISSUED INTO THE FPL SYSTEM.	RV	RH	TS	JCW

FPL

TURKEY POINT NUCLEAR UNIT 4

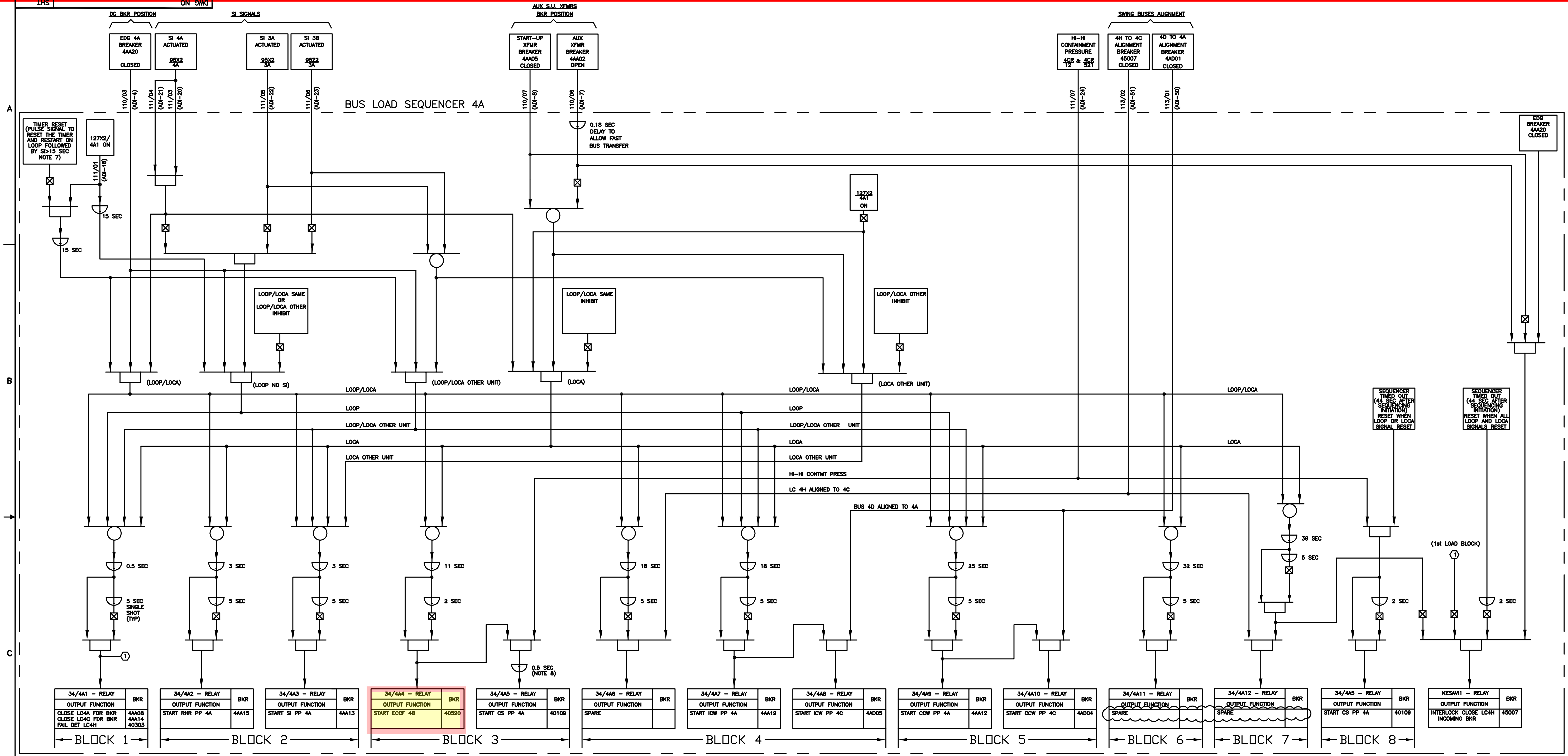
LOGIC DIAGRAM

EMERGENCY BUS LOAD SEQUENCER LOADING LOGIC DIAGRAM

FLORIDA POWER & LIGHT

DRAWING NUMBER: 5614-T-L1 | SYS: – | SHEET: 12B | REV: 3

5614-T-L1_SH 12B.DWG



BUS LOAD SEQUENCER 4A

***	LOAD CONDITIONS (EVENT)	1st LOAD BLOCK	2nd LOAD BLOCK		3rd LOAD BLOCK		4th LOAD BLOCK			5th LOAD BLOCK		6th LOAD BLOCK	7th LOAD BLOCK	8th LOAD BLOCK
1	LOOP	X						X	X	X	X			
2	LOOP WITH A SIMULTANEOUS LOCA SAME TRAIN (4A)	X	X	X	X		X	X	X	X	X	X	X	
3	LOOP FOLLOWED SOMETIME LATER BY A LOCA SAME TRAIN (4A)	X	X	X	X		X	X	X	X	X	X	X	
4	LOOP WITH A SIMULTANEOUS LOCA OTHER UNIT	X		X				X	X	X	X			
5	LOOP FOLLOWED SOMETIME LATER BY A LOCA OTHER UNIT	X		X				X	X	X	X			
6	LOCA SAME TRAIN (4A)	X	X	X	X		X	X	X	X	X	X	X	
7	LOCA OTHER UNIT			X										
8	HI-HI CONTAINMENT PRESSURE CONCURRENT OR LESS THAN 13 SECONDS AFTER A LOOP/LOCA	X	X	X	X	X	X	X	X	X	X	X	X	
9	HI-HI CONTAINMENT PRESSURE CONCURRENT OR LESS THAN 13 SECONDS AFTER A LOCA	X	X	X	X	X	X	X	X	X	X	X	X	
10	HI-HI CONTAINMENT PRESSURE OCCURING MORE THAN 13 SECONDS AFTER A LOCA	X	X	X	X		X	X	X	X	X	X	X	X
11	HI-HI CONTAINMENT PRESSURE OCCURING MORE THAN 13 SECONDS AFTER A LOOP/LOCA	X	X	X	X		X	X	X	X	X	X	X	X
12	LOCA SAME TRAIN FOLLOWED SOMETIME LATER BY LOOP	X	X	X	X		X	X	X	X	X	X	X	
13	LOCA OTHER UNIT FOLLOWED SOMETIME LATER BY LOOP	X		X				X	X	X	X			

LC 4H TRANSFER PERMISSIVE

THE KESAVI1 RELAY IS NORMALLY DE-ENERGIZED ALLOWING FOR LC-4H TRANSFER. THE RELAY ENERGIZES AS SOON AS THE AUX AND S/U XFMR BKRS ARE OPEN AND THE EDG 4A BREAKER CLOSES. IT THEN DE-ENERGIZES DURING THE FIRST LOAD BLOCK, ENERGIZES AFTER THE FIRST LOAD BLOCK, DE-ENERGIZES DURING THE SEVENTH LOAD BLOCK, ENERGIZES AFTER THE SEVENTH LOAD BLOCK, AND DE-ENERGIZES TWO SECONDS AFTER THE SEQUENCER HAS TIMED OUT.

NOTES:

1. THIS DRAWING REPRESENTS A CONCEPTUAL LOGIC DIAGRAM OF THE SEQUENCING LOADING FUNCTION OF THE EMERGENCY BUS LOAD SEQUENCER 4A.
2. DELETED
3. DELETED
4. THE EIGHTH LOAD BLOCK REPRESENTS A LATE START OF THE CONTAINMENT SPRAY PUMP DUE TO A LATE RECEIPT (TIME LATER THAN 13 SECONDS AFTER LOCA OR LOOP/LOCA) OF A HIGH-HIGH CONTAINMENT PRESSURE SIGNAL.
5. DELETED.
6. THE BLOCKS SHOWN INSIDE THE SEQUENCER BOX ARE GENERATED INTERNALLY BY THE SEQUENCER LOGIC.
7. ON LOOP FOLLOWED BY LOCA SCENARIO, THE TIMER IS RESET AND RESTARTS, IF SI OCCURS IN MORE THAN 15 SEC'S AFTER LOOP.
8. TIME DELAY PROVIDED TO ENSURE CONTAINMENT SPRAY PUMP START IF HI-HI CONTAINMENT PRESSURE IS RECEIVED NEAR END OF TWO SECOND LOAD BLOCK 3 PULSE.

NOTE: THIS DWG IS MADE FROM:
________ (VENDOR) DWG. NO. ________
________ (A/E) DWG. NO. ________ REV. ____
FPL DWG NO. 5610-T-L1, SH.12B REV.15

SAFETY RELATED | POD

REV	DATE	REVISION	BY	CH	APP	APP
3	2-16-13	ISSUED AS-BUILT PER EC 246931 (ITOP 13-02-097).	RV	BB	CJF	RP
2	10-21-96	ISSUED AS-BUILT PER PC/M 95-148.	RV	RH	BP	TS
1	3-17-96	ISSUED AS-BUILT PER PC/M 95-033.	RV	RH	–	TS
0	9-22-95	ISSUED AS-BUILT PER PC/M 95-032, INCORP CRN-I-3262 AND ISSUED INTO THE FPL SYSTEM.	RV	RH	TS	JCW

FPL

TURKEY POINT NUCLEAR UNIT 4
LOGIC DIAGRAM
EMERGENCY BUS LOAD SEQUENCER LOADING LOGIC DIAGRAM

FLORIDA POWER & LIGHT
DRAWING NUMBER 5614-T-L1
SYS –
REV 3
SHEET: 12A

5614-T-L1_SH 12A.DWG

Original for Question # 41.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	011	2.1.20	
	Importance Rating	4.6		

Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: RO Question # 41

Given the following conditions:

- Unit 3 Reactor trips and Safety Injection actuates.
- 3A 4KV Bus is de-energized due to an overcurrent condition.
- All RCPs are tripped in accordance with the Fold-Out page.
- The crew is performing Attachment 3 of 3-EOP-E-0, Reactor Trip or Safety Injection.
- All other safeguards equipment functions normally.

Which ONE of the following completes the following statement?

In accordance with Attachment 3 of 3-EOP-E-0, the RCO ___(1)___ manually start the Normal Containment Coolers.

The 3B ECC ___(2)___ automatically start.

A. (1) will
(2) will

B. (1) will
(2) will NOT

C. (1) will NOT

(2) will NOT

D. (1) will NOT

(2) will

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong. 2nd part right. Plausible to remember that two (2) NCCs must be running per Attachment 3. However, With the RCP not running, no NCC are started as that step is bypassed. Attachment 3 requires verification that ONLY two (2) ECCs are running.

B. **Incorrect**. 1st part wrong. 2nd part wrong. Plausible to remember that two (2) NCCs must be running per Attachment 3. However, With the RCP not running, no NCC are started as that step is bypassed. Attachment 3 requires verification that ONLY two (2) ECCs are running. Plausible to remember that the sequencer only starts the 3A and 3C ECCs and that 3B ECC does not get a sequencer start signal. However, since the 3C ECC did not start (loss of 3A 4KV Bus), the 3B ECC starts.

C. **Incorrect.** 1st part right. 2nd part wrong. Plausible to remember that the sequencer only starts the 3A and 3C ECCs and that 3B ECC does not get a sequencer start signal. However, since the 3C ECC did not start (loss of 3A 4KV Bus), the 3B ECC starts.

D. **Correct.** 1st part right. 2nd part right. Attachment 3 requires verification that ONLY two (2) ECCs are running. Since the 3C ECC did not start (loss of 3A 4KV Bus), the 3B ECC starts.

Technical Reference(s): 3-EOP-E-0 Rev 10. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: 6902332 Obj. 2, 3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

Question # 42

According to 3-EOP-E-1, Loss of Reactor or Secondary Coolant, whether or not containment spray may be terminated following a LOCA is based on __(1)__ and __(2)__.

A. (1) time since the event
(2) containment pressure

B. (1) time since the event
(2) containment temperature

C. (1) containment sump pH
(2) containment pressure

D. (1) containment sump pH
(2) containment temperature

Answer Analysis

A. Correct IAW 3-EOP-E-1.

B. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since adverse containment is based on temperature.

C. Incorrect.
Part 1 incorrect, plausible since the reason for the 30-day delay is to control sump pH, but that is not the criteria.
(2) Correct

D. (1) Incorrect, see C(1)
(2) Incorrect, see B(2)

Question Number: 42

Tier: 2 **Group:** 1

K/A: 026 (SF5 CSS) Containment Spray
026A4.05; Ability to manually operate and/or monitor in the control room:
Containment spray reset switches

Importance Rating: **3.5**

10 CFR Part 55: 41.7

10 CFR 55.43.b : XXX

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-EOP-E-1

Proposed references to be provided: None

Learning Objective: PTN 6902321 OBJ. 4

Cognitive Level: Higher _
Lower X_

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
➔ 12.	**Check If Containment Spray Should Be Stopped**	
	a. Containment Spray pumps – ANY RUNNING	**a.** Observe CAUTION prior to Step 13, and go to Step 13.
	b. Time since event – GREATER THAN OR EQUAL TO 30 DAYS	**b.** Perform the following: **1)** Check for indications of a LOCA: • Abnormal Containment radiation AND • Abnormal Containment pressure AND • Abnormal Containment Sump level **2)** IF a LOCA is indicated, THEN go to Step 13. **3)** IF a LOCA is **NOT** indicated, THEN perform Step 12.c.
	c. Containment pressure – LESS THAN 17 PSIG	**c.** WHEN Containment pressure is less than 17 psig, THEN perform Step 12.d, Step 12.e and Step 12.f. Observe CAUTION prior to Step 13 and continue with Step 13.
	d. Reset Containment Spray signal	
	e. Stop both Containment Spray pumps and place in standby	
	f. Close Containment Spray Isolation valves: • MOV-3-880A • MOV-3-880B	

Question # 43

Given the following:

- Unit 3 is at 100% power.
- An inadvertent closure of ONE MSIV occurs.

Which one of the following completes the statement below?

The RCS Cold Leg Temperatures on the two actively steaming loops will INITIALLY __(1)__.

An AUTOMATIC Safety Injection is expected to be triggered FIRST by the __(2)__ SI logic.

A. (1) rise
(2) High Steam Line Differential Pressure

B. (1) rise
(2) High Steam Flow With Low SG Pressure or Low Tave

C. (1) lower
(2) High Steam Line Differential Pressure

D. (1) lower
(2) High Steam Flow With Low SG Pressure or Low Tave

Answer Analysis .

Discussion: When one MSIV closes at RTP, the other two SGs steam at a higher rate. This increase in steam flow is enough to trip the high steam flow bistables (114.4%) for the two active Steam lines. When the steaming rate increases, their pressures drop. When the active SG pressures reach 614#, an SI occurs. Since the steam header setpoint has a floor of 585#, the Delta P SI will occur at a pressure of 485#. This is a lower pressure, so it will not be reached first.

A. Incorrect.
   Part 1 incorrect, plausible since the loop with the closed MSIV will have a rising cold leg temperature.
   Part 2 incorrect, plausible since this SI is also challenged, but it has a lower setpoint and will not be reached first.

B. Incorrect.
   Part 1 incorrect, plausible since the loop with the closed MSIV will have a rising cold leg temperature.
   Part 2 incorrect, plausible since this SI is also challenged, but it has a lower setpoint and will not be reached first.

C. Incorrect.
   Part 1 correct.
   Part 2 incorrect, plausible since Plausible because this SI is also challenged, but it has a lower setpoint and will not be reached first.

D. Correct, see discussion above

Question Number: 43

Tier: **2** **Group:** 1

K/A: **039K3.05**
Knowledge of the effect that the loss of MRSS will have on the RCS.
(**MRSS is Main and Reheat Steam**)

Importance Rating: **3.6**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions

SRO Justification: N/A

Technical References: 5610-T-L1 SH 19, 5613-M-3072 SH 1.
Lesson Plan 6910117 -- Main and Extraction Steam Systems
Lesson Plan 6910163 - RPS

Proposed references to be provided: None

Learning Objective: PTN 6902163 OBJ. 6

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

PTN 6910117 MAIN AND EXTRACTION STEAM SYSTEM

Inadvertent MSIV Closure

The transient produced by the inadvertent closure of a MSIV would produce a Reactor Trip and place the plant in the EOP network. Adequate heat removal capability is available to the primary system with the MSIVs open or closed.

Engineered Safety Feature Actuation System Trip

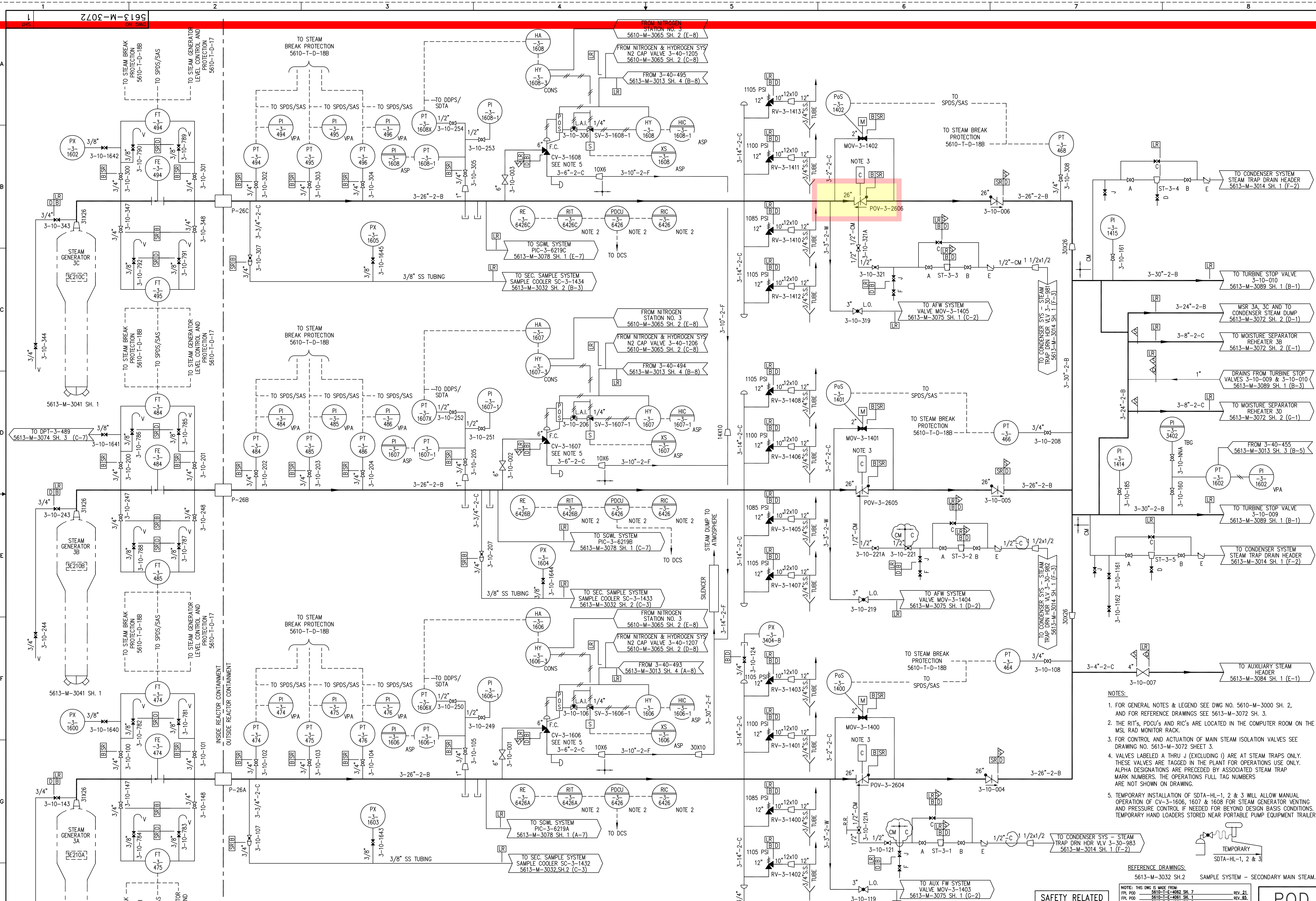
A reactor trip will occur upon a safety injection actuation of the Engineering Safety Features Actuation System (ESFAS). See Figure 15. Table 2 lists the actuation setpoints for the various safety injection signals.

The means of actuating the Engineering Safety Features Actuation System and tripping the reactor are as follows:

1. Low pressurizer pressure trip at 1730 PSIG decreasing. If any two of the three pressure transmitter circuits lowers to the SI setpoint, the result will be a SI and a reactor trip. This trip may be manually blocked below 2000 PSIG.
2. High steam line differential pressure between any one steam generator and steam header can also result in a SI and a reactor trip. The setpoint is 100 PSID on two of the three of SG pressure transmitters on one steam generator compared to steam header pressure. The steam header pressure transmitters do not go below 585 psig to protect from a steam line break when the MSIVs are closed. This ESF actuation trip may be manually blocked below 2000 PSIG.
3. High steam flow coincident with either low steam generator pressure or low T_{AVG}. One out of two high steamline flow signals in two out of three of the steam generators, with either a low T_{AVG} of 543°F in two out of three loops, or a low steam generator pressure of 614 PSIG in two out of three steam generators, will result in

ESF actuation and a reactor trip. This circuit may be manually blocked below 543°F to allow a normal plant shutdown

MSIVs stay operable but closed. So TS 3.7.1.5 is met



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2, AND FOR REFERENCE DRAWINGS SEE 5613-M-3072 SH. 3.
2. THE RIT's, PDCU's AND RIC's ARE LOCATED IN THE COMPUTER ROOM ON THE MSL RAD MONITOR RACK.
3. FOR CONTROL AND ACTUATION OF MAIN STEAM ISOLATION VALVES SEE DRAWING NO. 5613-M-3072 SHEET 3.
4. VALVES LABELED A THRU J (EXCLUDING I) ARE AT STEAM TRAPS ONLY. THESE VALVES ARE TAGGED IN THE PLANT FOR OPERATIONS USE ONLY. ALPHA DESIGNATIONS ARE PRECEDED BY ASSOCIATED STEAM TRAP MARK NUMBERS. THE OPERATIONS FULL TAG NUMBERS ARE NOT SHOWN ON DRAWING.
5. TEMPORARY INSTALLATION OF SDTA-HL-1, 2 & 3 WILL ALLOW MANUAL OPERATION OF CV-3-1606, 1607 & 1608 FOR STEAM GENERATOR VENTING AND PRESSURE CONTROL IF NEEDED FOR BEYOND DESIGN BASIS CONDITIONS. TEMPORARY HAND LOADERS STORED NEAR PORTABLE PUMP EQUIPMENT TRAILER.

REFERENCE DRAWINGS:

5613-M-3032 SH.2 SAMPLE SYSTEM - SECONDARY MAIN STEAM.

SAFETY RELATED

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4062 SH. 7 REV. 21
FPL POD 5610-T-E-4061 SH. 1 REV. 65
FPL POD 5610-T-E-4061 SH. 3 REV. 36

POD

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
37	10-4-11	ISSUED AS-BUILT PER EC-DCR 273958.	RH	BB	BSG	VMG	41	7-25-12	ISSUED AS-BUILT PER EC 242385 (PC/M 08-017).	RV	BB	DM	CM
46	6-6-17	ISSUED AS-BUILT PER EC 242385.	AFG	RH	–	BB	40	7-9-12	ISSUED AS-BUILT PER EC 246875.	RH	RV	PB	JD
45	7-15-13	ISSUED AS-BUILT PER EC 250185 (ITOP 13-07-032).	RH	RV	–	EB	39	06-29-12	ISSUED AS-BUILT PER EC 246875. (PARTIAL)	RV	RH	–	SB
44	2-6-13	ISSUED AS-BUILT PER EC 271032 (ITOP 13-02-001).	RV	AFG	BB	BAP	38	06-05-12	ISSUED AS-BUILT PER EC 246875. (PARTIAL)	RV	SB	–	CW
43	12-2-12	ISSUED AS-BUILT PER CRNs 007 & 011 (EC 242400). ITOP 12-12-003.	AFG	PSB	–	JD	0	1-16-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	TAM	MD	LRB/RAS	HAR
42	11-16-12	ISSUED AS-BUILT PER EC-DCR 250231.	RH	JPH	–	SB	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-90-512.	—	—	—	—

FPL

TURKEY POINT NUCLEAR UNIT 3

P & ID

MAIN STEAM SYSTEM

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3072

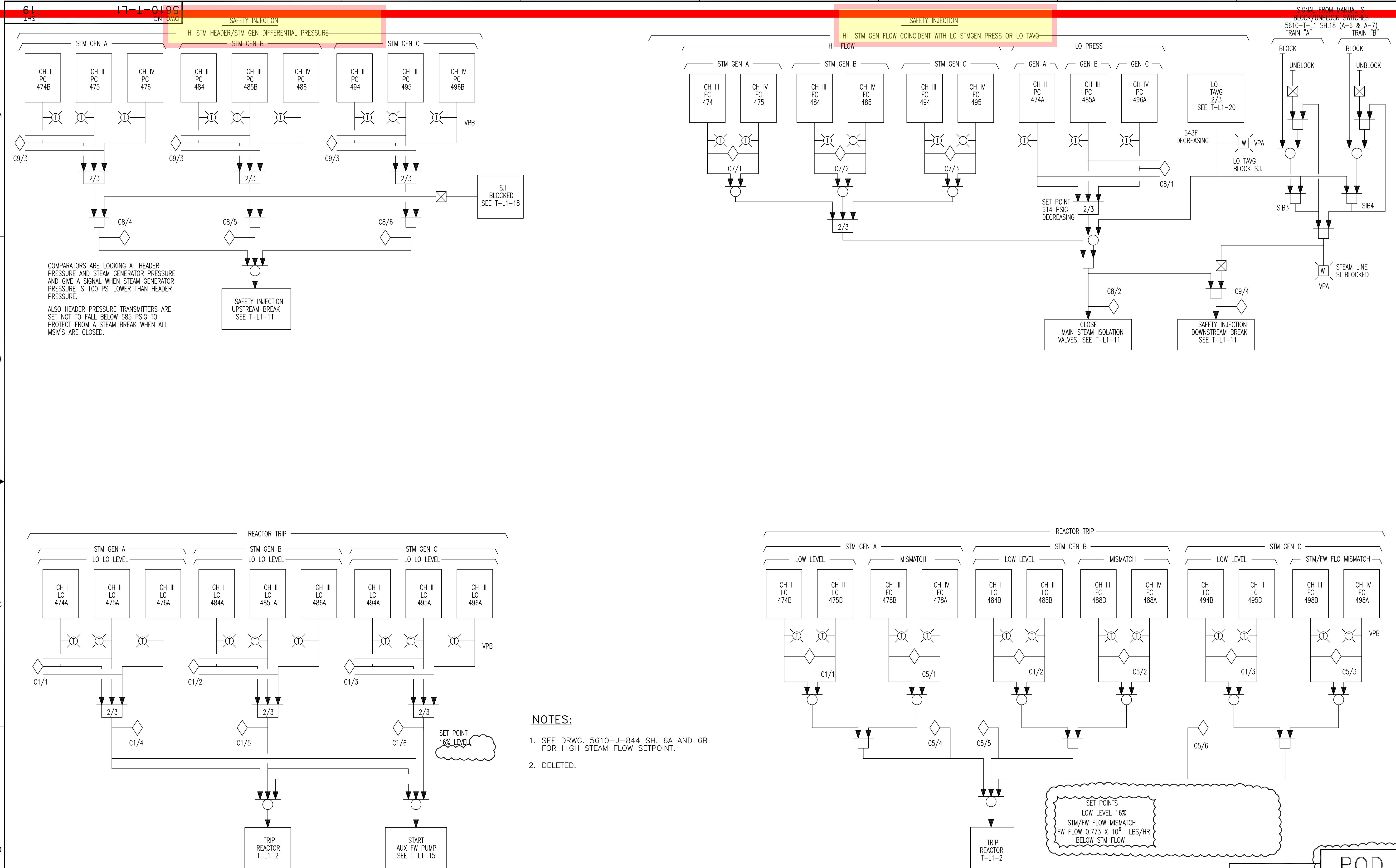
SYS 072

SHEET 1

REV 46

30x42 E SIZE 02/26/93

5613-M-3072_SH 1.DWG



NOTES:

1. SEE DRWG. 5610-J-844 SH. 6A AND 6B FOR HIGH STEAM FLOW SETPOINT.
2. DELETED.

SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP
19	10-28-96	ISSUED AS-BUILT FOR PC/M 95-171. (PARTIAL)	RV	RH	JG	JJN
18	10-17-96	ISSUED AS-BUILT FOR PC/M 95-171 (PARTIAL) & INCORP. CRN-I-3465.	RV	RH	JG	TS
17	10-9-96	ISSUED AS-BUILT PER PC/M 95-170. (PARTIAL)	RV	RH	–	TS
26	5/1/13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-04-155).	RV	BB	–	PMB
25	8-9-12	ISSUED AS-BUILT PER CRN-006 (EC 247048).	RV	RH	–	PMB

REV	DATE	REVISION	BY	CH	APP	APP
24	7-17-12	ISSUED AS-BUILT PER EC 247048 (PC/M 10-039).	RV	BB	–	PMB
23	07-31-01	ISSUED AS-BUILT PER CRN-I-3967 (PC/M 00-016).	RV	RH	BP	BAP
22	1-20-98	ISSUED AS-BUILT PER CRN-I-3663 (PC/M 97-042).	RV	RH	MA	TS
21	1-21-97	ISSUED AS-BUILT FOR PC/M 95-171.	RV	RH	JG	TS
20	1-13-97	ISSUED AS-BUILT FOR PC/M 95-170.	RV	RH	JG	TS

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

LOGIC DIAGRAM

STEAM GENERATOR CAUSED REACTOR TRIP AND SAFETY INJECTION SIGNALS

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-L1

SYS –

REV 26

SHEET: 19

5610-T-L1_SH 19.DWG

Question # 44

The crew is performing warmup of MSRs in accordance with 3-NOP-072.01, Moisture Separator Reheaters.

3-NOP-72.01 directs that for a MANUAL warmup of MSRs, the heatup rate be controlled using __(1)__.
An AUTOMATIC MSR warmup will be controlled using __(2)__.

A. (1) the applicable DCS screen
(2) the applicable DCS screen

B. (1) the applicable DCS screen
(2) an M/A station mounted on the control boards

C. (1) local action
(2) the applicable DCS screen

D. (1) local action
(2) an M/A station mounted on the control boards

Answer Analysis

Both manual and automatic MSR heatup are performed on the applicable DCS screen.

A. (1) Correct
(2) Correct

B. (1) Correct
(2) Incorrect – Distrators are plausible because they require plant specific knowledge of control locations

C. (1) Incorrect
(2) Correct

D. (1) Incorrect
(2) Incorrect

Question Number: 44

Tier: 2 **Group:** 1

K/A: 039 (SF4S MSS) Main and Reheat Steam
039G2.1.30; Ability to locate and operate components, including local controls.

Importance Rating: **4.4**

10 CFR Part 55: 41.7

10 CFR 55.43.b : XXX

K/A Match:

SRO Justification: N/A

Technical References: 3-NOP-72.01

Proposed references to be provided: None

Learning Objective: PTN 6902407 OBJ. 4

Cognitive Level: Higher _
Lower X

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

4.1.2 Placing Moisture Separator Reheaters in Service

1. **ENSURE** MSRs have been purged for a minimum of 15 minutes per Section 4.1.1.

2. IF desired to perform Manual Warmup of MSRs, THEN **GO TO** Section 5.10, MSR Manual Warmup

NOTE

- MSR temperature ramp rate can be adjusted between 0 and 50°F/Hr.
- Unless noted otherwise, the actions of this section will be performed on MSR WARM-UP/COOL-DOWN INTERFACE DCS screen, shown on Attachment 5, MSR Warm-Up/Cool-Down Interface.

3. **ADJUST** MSR temperature ramp rate to desired value.

NOTE

Program in DCS will automatically open the timing valves to 11% OPEN, hold that position for a 30 minute soak, and then ramp MSR outlet temperature until timing valves are fully open.

4. **SELECT** START MSR AUTO WARM-UP on DCS.

5. **ENSURE** M/A station in AUTO and timing valves start to open.

NOTE

The rate of change of crossover steam temperature out of the MSR is limited to less than 100°F/hr.

6. **MONITOR** CONTROL TEMP ROC DEGF/HR and valve operation during warmup.

7. IF any MSR timing valve is **NOT** opening, THEN **PERFORM** Section 5.5.1 for affected MSR.

5.10 MSR Manual Warmup

NOTE

- The MSRs are usually warmed up in Auto per Section 4.1.2. This section is intended for use only when directed by the Shift Manager.
- Unless noted otherwise, the actions of this section will be performed on MSR WARM-UP/COOL-DOWN INTERFACE DCS screen, shown on Attachment 5, MSR Warm-Up/Cool-Down Interface.

CAUTION

- If during performance of this section, the manual opening of MSR Timing Valves results in degradation of normal plant parameters, then MSR Timing Valves should be closed as necessary to restore plant parameters to normal.
- If possible, the time between opening each of the MSR Timing Valves should be performed with minimum delay.

1. **OBTAIN** permission from Shift Manager to perform MSR Manual Warmup.
2. Manually **POSITION** CV-3-3710, 3A MSR TIMING VLV to 5% OPEN using DCS as follows:
   - A. **PLACE** M/A STATION for CV-3-3710 in MANUAL.
   - B. **ADJUST** CV-3-3710 to 5% OPEN.
   - C. IF directed by Shift Manager, THEN **ADJUST** CV-3-3710 as necessary to restore plant parameters to normal.
3. Manually **POSITION** CV-3-3711, 3B MSR TIMING VLV, to 5% OPEN using DCS as follows:
   - A. **PLACE** M/A STATION for CV-3-3711 in MANUAL.
   - B. **ADJUST** CV-3-3711 to 5% OPEN.
   - C. IF directed by Shift Manager, THEN **ADJUST** CV-3-3711 as necessary to restore plant parameters to normal.

Question # 45

Given the following:

- 3-OSP-023.2, Diesel Generator 3A, 24 Hour Full Load Test and Load Rejection is in progress.

Which one of the following completes the statements below?

The 100% continuous steady state loading on the 3A EDG is allowed up to a MAXIMUM of __(1)__.

During the 110% loading portion of the test with EDG conditions being: Voltage 4160V, Current 430A, Load 2700kW, the RO __(2)__ required to adjust 3A EDG amperage.

REFERENCE PROVIDED

A. (1) 2500 kW
(2) is

B. (1) 2500 kW
(2) is NOT

C. (1) 2750 kW
(2) is

D. (1) 2750 kW
(2) is NOT

Answer Analysis

Discussion: Correct answer is A. Unit 3 EDG continuous load rating is 2500 KW.

3-OSP-023.2 tests the overload and 2-hour capability. EDG amperes must be controlled during the run to avoid excess current vs load and voltage. A curve and/or table in Enclosure 1, each, show the upper and lower limits for the voltage vs load. If amps are below the “lower” limit as described in the question stem then the voltage must be raised. (TO achieve an amperage between 444 – 477A)

A. Correct, see discussion above

B. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible if applicant reads curve or stem incorrectly. Also because raising voltage for a problem may seem counter intuitive

C. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible if applicant reads curve or stem incorrectly. Also because raising voltage for a problem may seem counter intuitive

D. Incorrect.
Part 1 incorrect, plausible since this number is less than the continuous loading rating for Unit 4 EDGS.
Part 2 incorrect, plausible if applicant reads curve or stem incorrectly. Also because raising voltage for a problem may seem counter intuitive

Question Number: 45

Tier: 2 **Group:** 1

K/A: 064K4.04: Knowledge of ED/G system design feature(s) and/or interlock (s) which provide for the following. Overload Ratings

Importance Rating: **3.1**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: **RO system question.** The K/A is matched as the applicant must demonstrate knowledge of EDG overload ratings and use OSP curve to prevent overload conditions

SRO Justification: N/A

Technical References: Lesson Plan PTN 69136, Emergency Diesel Generators
3-OSP-023.2 , **Diesel Generator 24 Hour Full Load Test and Load Rejection**

Proposed references to be provided: **YES,** 3-osp-023.2 Enclosure 1.
Learning Objective: PTN 6902136 OBJ. 4

Cognitive Level: Higher X
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

2.8 GENERATOR

2.8.1 Ratings

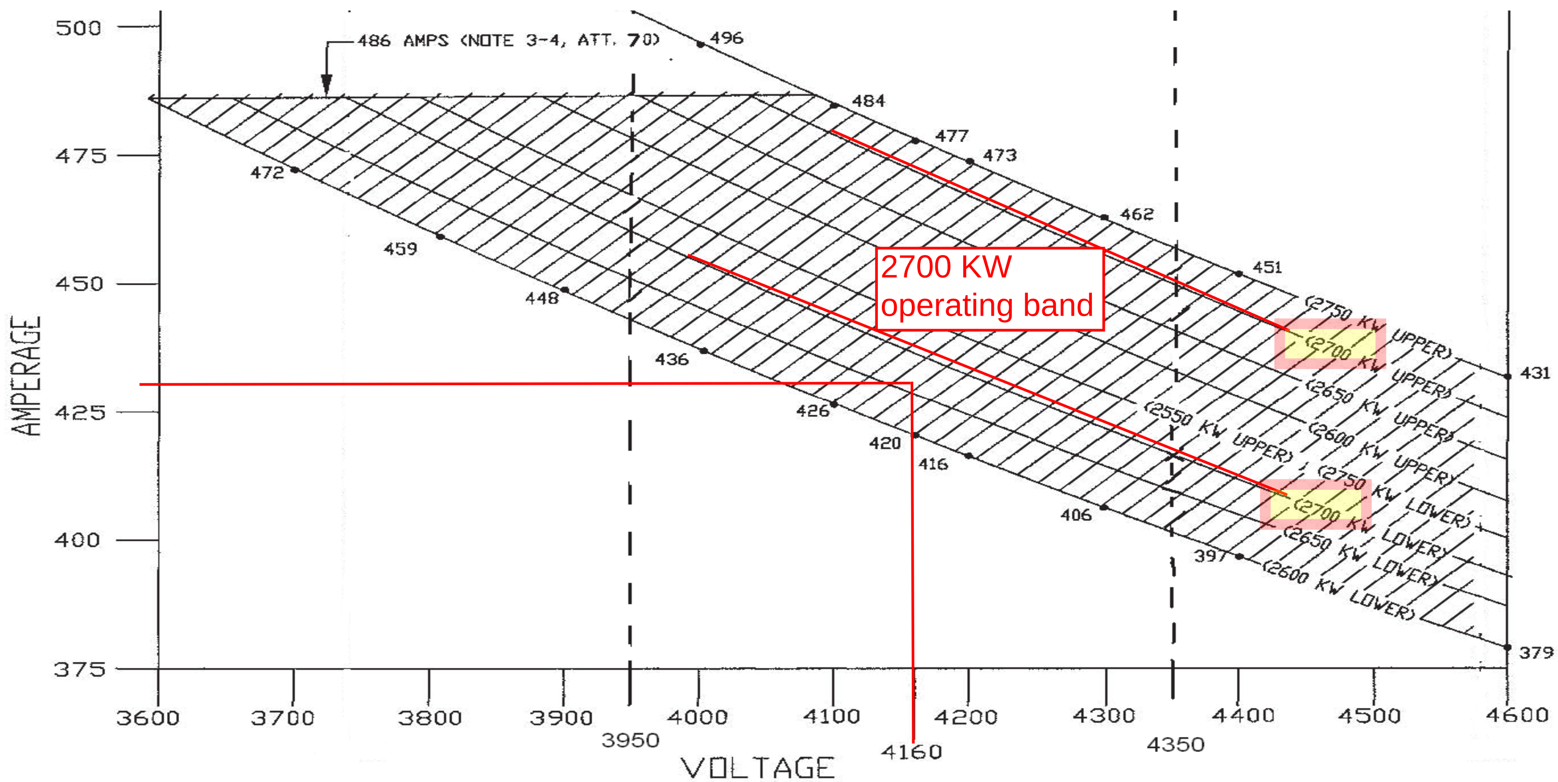
The generator is a 4160 volt, 3 phase, 60 Hz, AC synchronous machine. Its net output (at a 0.8 pf) is rated as follows:

FUNCTION	UNIT 3	UNIT 4
Base Continuous (24 hrs/ 365 days/ yr)	2500 KW	2874 KW
Basic Overload (2 hr/ every 24 hr	2750 KW	3162 KW
Peak Rating (2000 hrs per year)	2850 KW	3095 KW (orange mark)
200 hr rating	None	3237 KW (red mark)
168 hr rating (7 day)	2950 KW	None
4 hour rating	None	3266 KW
½ hour rating	3050 KW	3295 KW

ENCLOSURE 1
(Page 4 of 8)

UNIT 3 EDG 100% AND 110% OPERATING RANGES

UNIT 3 110%, SURVEILLANCE OPERATING RANGE



Acceptable Operating Range is such that the intersection of Voltage and Amperage are within the Operating kW Band. At all times, the equation below may be used to generate the Lower and Upper current limits needed to be maintained given the present EDG voltage.

- Enter the graph with voltage, and find the intersection with current. This point is required to be within the shaded region for the present EDG kW loading. Examples are in the following Note section.
- If EDG loading (kW) is between the lines, the upper and lower current limits (Amperage) can be calculated with the following equation:

$$\text{Amps} = \frac{(\text{kW})(1000)}{(E)(pf)(1.732)}$$

Lower Limit: kW = present loading, E = present voltage, pf = 0.86
Upper Limit: kW = present loading, E = present loading, pf = 0.80

Question # 46

Given the following:

- Unit 3 experiences a 3A SG Fault from 100% power.
- 3-EOP-FR-H.1, Response to Loss of Secondary Heat Sink, is in progress.
- 3A SG WR Level is 0%.
- 3B SG NR Level is 10%.
- 3C SG NR Level is 10%.
- Adverse Containment conditions exist.

Which one of the following completes the statements below?

The crew will establish Standby Feedwater Flow to __(1)__ SG(s) at __(2)__.

A. (1) ONLY 3B and 3C
(2) a MAXIMUM rate of 100 gpm per SG

B. (1) ONLY 3B and 3C
(2) at a rate with no MAXIMUM limit per SG

C. (1) ALL
(2) a MAXIMUM rate of 100 gpm per SG

D. (1) ALL
(2) at a rate with no MAXIMUM limit per SG

Answer Analysis

With adverse containment, NR of 25% is a dry SG.

A. Correct. Feed Flow is ONLY established to the intact SGs. With adverse containment, NR of <27% requires to limit feed flow below 100 gpm.

B. (1) Part 1 correct.
(2) Part 2 incorrect, plausible since for normal containment conditions the given NR Level values for the intact SGs would not require any flow limitations.

C. (1) Part 1 incorrect, plausible since for a Loss of Heatsink Scenario with no SG fault, feeding all SGs would be correct.
(2) Part 2 correct.

D. (1) Part 1 incorrect, plausible since for a Loss of Heatsink Scenario with no SG fault, feeding all SGs would be correct.
(2) Part 2 incorrect, plausible since for normal containment conditions the given NR Level values for the intact SGs would not require any flow limitations.

Question Number: 46

Tier: 2 **Group:** 1

K/A: 059 (SF4S MFW) Main Feedwater
059A2.04 ; Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feeding a dry S/G

Importance Rating: **2.9**

10 CFR Part 55: 41.5

10 CFR 55.43.b **:** XXX

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-EOP-FR-H.1

Proposed references to be provided: None

Learning Objective: PTN 6902337 OBJ. 4

Cognitive Level: Higher X _
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

CAUTION

- If CST level decreases to less than 12%, makeup water sources for CST will be necessary to maintain secondary heat sink.
- Feed flow is required to be initiated at a rate **NOT** to exceed 100 gpm to avoid excessive RCS cooldown and to limit thermal stress in the S/G(s) with Wide Range Level less than 9%[Narrow Range less than 27%]. This 100 gpm feed flow limit applies to each individual S/G and only while that S/G Wide Range Level is less than 9%[Narrow Range less than 27%].

3. Try To Establish AFW Flow To At Least One S/G

a. Check S/G Blowdown Isolation:

- S/G Blowdown Isolation Valves – CLOSED
- S/G Sample Isolation Valves – CLOSED

RNO: **a.** Manually close valves.

b. Check Control Room indications for cause of AFW failure:

- CST level
- AFW Steam Supply MOV power supply
- AFW valve alignment

c. Try to restore AFW flow

Question # 47

Which one of the following completes the statements below?

At 100% power, MOV-3-1403, SG A Stm Supply To AFW Pumps, is NORMALLY powered by ______.

A. 3D01, 3A Vital DC Bus

B. 3D23, 3B Vital DC Bus

C. 4D01, 4B Vital DC Bus

D. 4D23, 4A Vital DC Bus

Answer Analysis

A. Incorrect, plausible since this is the power supply to other AFW components, i.e. MOV-3-1405.

B. Incorrect, plausible since this is a Vital DC power supply to other AFW components, i.e. Unit 3 Train 2 FCVs.

C. Correct. 3(4) -MOV – 1403 are both powered from DC bus 4D01

D. Incorrect, plausible since this is a Vital DC power supply to other AFW components, i.e. Unit 4 Train 1 FCVs.

Question Number: 47

Tier: 2 **Group:** 1

K/A: **061AK2.01**
Knowledge of bus power supplies to AFW system MOVs

Importance Rating: **3.2**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must demonstrate knowledge of vital power supplies, specifically the AFW MOVs. Shows understanding of redundancy and independent power supplies.

SRO Justification: N/A

Technical References: 5610-E-855

Proposed references to be provided: None

Learning Objective: PTN 6910123 OBJ. 4

Cognitive Level: Higher _

Lower X Fundamental knowledge

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable
Lesson Plan 6910123

INPUT DATA FILE PTP.BL.98

4D01 (125V DC DISTRIB. SWGR.)

CODE	HEATER BREAKER	SCHEMATIC LOCATION PNL SCHED.	EQUIP. NO. CKT DESC	EQUIPMENT AND OR LOCATION	BREAKER NUMBER
202		E-11	4C264	ALTERNATE SHUTDOWN PANEL (DCR-89-144)	4D01-25
	4P82-04	E-11	MOV4-1403	D.C. MOV 1403 (5177-136-E-05.SH.1) PCM 80-798	4D01-26
25		E-11	3AB	4160V SWGR 3B BUS DUP-064	4D01-27
7	3P82-04	E-11	MOV3-1403	DC. MOV 1403(5177-136-E-05, SH.1) (PCM 80-78) VIA STARTER 3N1403	4D01-28
403		E-11 E-28 SH.30B	4C23B 4QR45	RCP UNDERFREQUENCY TRIPS (4C23B) AND SFGDS RACK 4QR45 NOTE: SEE BKRS 4P08-12 (CRN-E-15591 PC/M 96-064)	4D01-29
298		E-11	4C31	4A ROD DRIVE MG FLASHING & CONTROL POWER NOTE: SEE BKR 4D01-34 (DCR-TPE-92-079)	4D01-30
25		127-E-05-SH1	4C03	SOLENOID OPERATED CONTAINMENT ISOLATION VALVE IN GAS ANALYZER 4C03 (PCM 80-52) DUP-064	4D01-31
25		E-11	RTU	REMOTE TERMINAL UNIT #4 DUP-064	4D01-32
25		E-11	4D06	MISC VALVES DC DISTRIBUTION PNL 4D06 (3V06 & 40L) NOTE: SEE BKRS 4P09-18,3P09-18 DUP-064	4D01-33
25		E-11	4C32	4B ROD DRIVE MG FLASHING & CONTROL POWER NOTE: SEE BKR 4D01-30 DUP-064	4D01-34
4		E-11		G.A. CTMT. ISOL. VLV. PNL 3C03 (PC/M 80-51)	4D01-35
25		E-11	C25 C38	DRUMMING RM PNL C38 & WB PNL C25 NOTE:SEE BKRS 3D01-23,3P06-7,LP311-38,LP311A-33 DUP-064	4D01-36

CODES: NAB-NOT AS BUILT
RES-RESERVED

Question # 48

Which one of the following is the LOWEST SG pressure that is capable of supporting the operation of the AFW Pumps for feeding the SGs?

A. 450 psig

B. 360 psig

C. 220 psig

D. 150 psig

Answer Analysis

Discussion: Correct answer D. AFW turbines can pump water to the SGs down to 85psig.

A. Incorrect. Plausible since this is a set point use in the EOPs (limit for placing RHR in service), and is lower than Normal SG pressure.

B. Incorrect. Incorrect. Plausible since this is a set point use in 3-EOP-FR-H.1 (limit for SG depressurization in order to feed SGs from a Condensate Pump) and is lower than Normal SG pressure.

C. Incorrect. Plausible since this is the one of the stopping points for SG depressurization in 3-EOP-E-3, and it is plausible that it is required to stop at this point in order to continue to feed the SGs on AFW.

D. Correct: See the above discussion

Question Number: 48

Tier: 2 **Group:** 1

K/A: 061 (SF4S AFW) Auxiliary/Emergency Feedwater
061AA1.02: Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the AFW controls including:
S/G pressure

Importance Rating: **3.3**

10 CFR Part 55: 41.5

10 CFR 55.43.b : XXX

K/A Match: The K/A is matched as the applicant must assess the ECA 0.0 purpose for reducing steam generator pressures only to 150 psig and evaluate AFW design capability to feed SGs . Then the RO applicant must understand the purpose of waiting to align RHR until less than 350 degrees on RCS Thot.

SRO Justification: N/A

Technical References: PTN 6910123

Proposed references to be provided: None

Learning Objective: PTN 6910123 OBJ. 6

Cognitive Level: Higher _
Lower X_

Question Source: New X_
Modified Bank _
Bank

Question History: New

Comments: Screenshot of development references as applicable

AFW - Turbine

- Single wheel-helical flow
- 900 HP @ 5900 RPM with inlet steam pressure of 985 PSIG
- The FSAR shows a **minimum operating steam pressure of 85 PSIG.**
- ***Below 85 PSIG, the turbine begins to consume more mass than the AFW pump can deliver to the S/Gs. Above 85 PSIG the AFW pump delivers more mass than is consumed by the turbine.***

NEXTera ENERGY NUCLEAR FLEET

OBJECTIVE 6.d

Question # 49

- Both Units are at 100% power.

Subsequently:

- 3B MCC Vital, DE-ENERGIZES.

Which one of the following identifies the impact on the Unit 3 Vital DC buses?

A. Unit 3 Vital DC busses remain un-affected.

B. ONLY the 3D01, 3A Vital DC Bus, loses one Battery Charger

C. ONLY the 3D23, 3B Vital DC Bus, loses one Battery Charger

D. BOTH the 3A and 3B Vital DC busses lose one Battery Charger each

Answer Analysis

A. Incorrect, plausible since some DC Buses remain unaffected, i.e. 4D01 and 4D23.

B. Incorrect, plausible since the battery chargers for this Vital DC Bus are powered from other Vital MCCs.

C. Correct. 3B MCC Vital powers the 3B1 battery charger to the 3B Vital DC Bus.

D. Incorrect, plausible since the loss of some Vital MCCs will cause multiple battery chargers to be lost, i.e. Loss of 4D Vital MCC will cause power to be lost to the 3A2 and 3B2 Battery Chargers.

Question Number: 49

Tier: **2** **Group:** 1

K/A: **062AK3.02**
Knowledge of the effect of a loss of AC power on the DC power system

Importance Rating: **3.7**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: **RO question addressing the PURPOSE of the action to initiate Phase A on a loss of DC.** The K/A is matched as the applicant must evaluate plant conditions associated with a loss of all vital power and impact on the DC power systems. The impact of the loss of all AC is that (among other actions) the Containment Isolation racks must be deenergized to minimize DC loads and ensure isolation prior to losing DC power

SRO Justification: N/A

Technical References: 5610-T-E-1592 SH 1

Proposed references to be provided: None

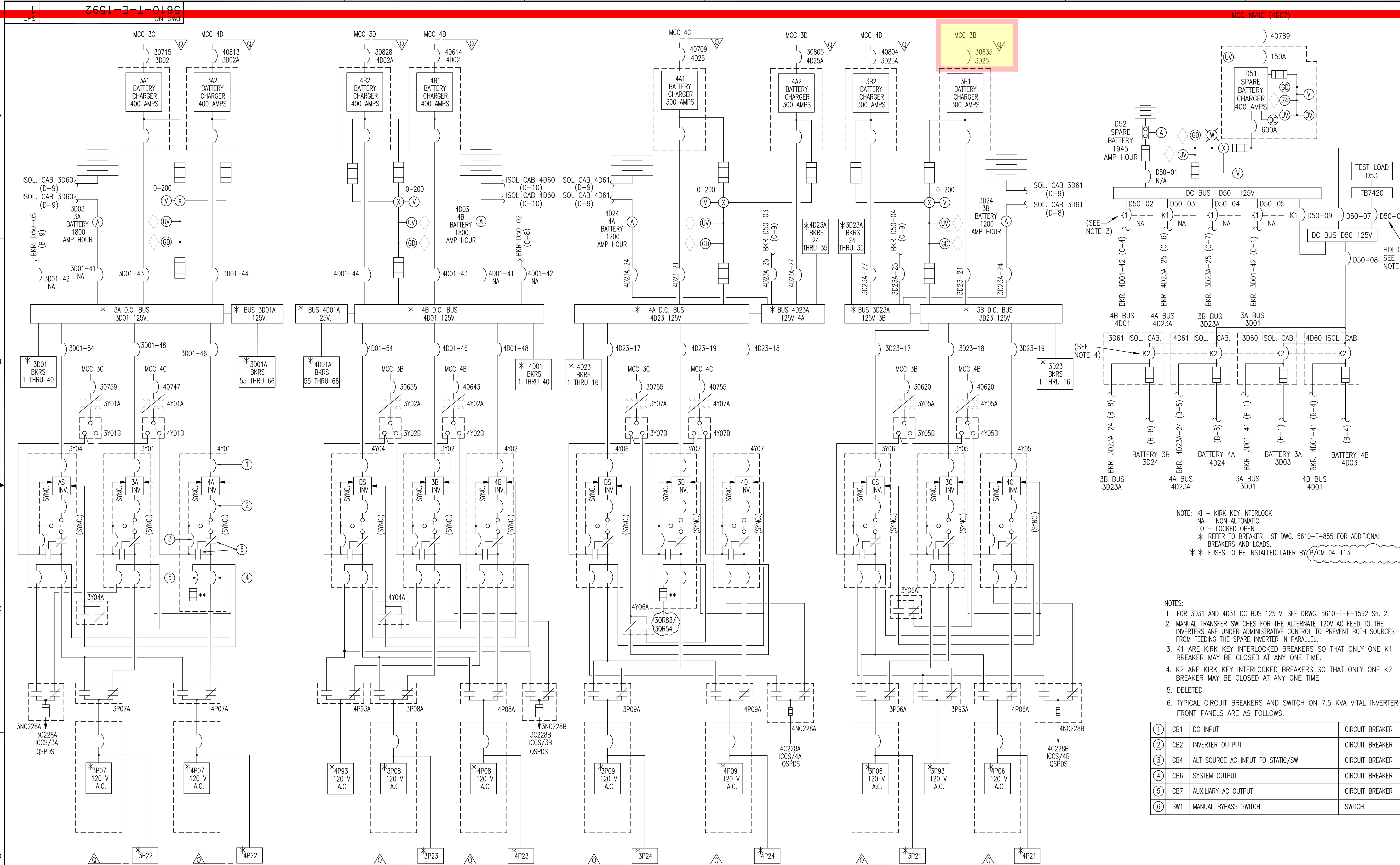
Learning Objective: PTN 6902139 OBJ. 4

Cognitive Level: Higher __
Lower _X

Question Source: New
Modified Bank X_
Bank _

Question History: Modified question #48 from PTN 2018 exam.

Comments: Screenshot of development references as applicable



NOTE: KI – KIRK KEY INTERLOCK
NA – NON AUTOMATIC
LO – LOCKED OPEN
∗ REFER TO BREAKER LIST DWG. 5610–E–855 FOR ADDITIONAL BREAKERS AND LOADS.
∗∗ FUSES TO BE INSTALLED LATER BY P/CM 04–113.

NOTES:

1. FOR 3D31 AND 4D31 DC BUS 125 V. SEE DRWG. 5610–T–E–1592 Sh. 2.
2. MANUAL TRANSFER SWITCHES FOR THE ALTERNATE 120V AC FEED TO THE INVERTERS ARE UNDER ADMINISTRATIVE CONTROL TO PREVENT BOTH SOURCES FROM FEEDING THE SPARE INVERTER IN PARALLEL.
3. K1 ARE KIRK KEY INTERLOCKED BREAKERS SO THAT ONLY ONE K1 BREAKER MAY BE CLOSED AT ANY ONE TIME.
4. K2 ARE KIRK KEY INTERLOCKED BREAKERS SO THAT ONLY ONE K2 BREAKER MAY BE CLOSED AT ANY ONE TIME.
5. DELETED
6. TYPICAL CIRCUIT BREAKERS AND SWITCH ON 7.5 KVA VITAL INVERTER FRONT PANELS ARE AS FOLLOWS.

1	CB1	DC INPUT	CIRCUIT BREAKER
2	CB2	INVERTER OUTPUT	CIRCUIT BREAKER
3	CB4	ALT SOURCE AC INPUT TO STATIC/SW	CIRCUIT BREAKER
4	CB6	SYSTEM OUTPUT	CIRCUIT BREAKER
5	CB7	AUXILIARY AC OUTPUT	CIRCUIT BREAKER
6	SW1	MANUAL BYPASS SWITCH	SWITCH

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
45	12–18–12	ISSUED AS–BUILT PER EC 242033. (ITOP 12–12–020)	RV	BB	–	BAP	40	07–13–04	ISSUED AS–BUILT PER CRN–E–16624 (PC/M 03–106).	RH	RV	FJK	TS
44	12/22/08	ISSUED AS–BUILT PER PC/M 04–048.	RV	BB	–	PMB	39	07–31–01	ISSUED AS–BUILT PER CRN–E–16236 (PC/M 00–016).	RV	RH	MCP	TAB
43	11/24/08	ISSUED AS–BUILT PER PC/M 04–048. (PARTIAL)	RV	DAR	–	PMB	38	10–21–97	ISSUED AS–BUILT PER CRN'S E–15712 & E–15708(PC/M 97–028).	RV	RH	FJK	RB
42	7/25/07	ISSUED AS–BUILT PER PC/M 03–109.	RV	RH	–	PDS	37	4–25–94	INCORP. CRN–E–13300 FOR PC/M 93–047.	RV	RH	BAW	JCW
41	5/3/06	ISSUED AS–BUILT PER PC/M 03–135.	RV	RH	FJK	PMB	36	7/29/91	ISSUED AS–BUILT FOR PC/M's 83–06 & 87–265.	RH	LH	–	JG

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

OPERATING DIAGRAM

125V D.C. & 120V INSTRUMENT A.C. ELECTRICAL DISTRIBUTION

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610–T–E–1592

SYS –

SHEET: 1

REV 45

5610–T–E–1592_SH 1.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	063		K1.03
	Importance Rating	2.9		

Knowledge of the physical connections and/or cause-effect relationships between the dc electrical system and the following systems: Battery charger and battery

Proposed Question: RO Question # 48

Given the following:

- Both units are at 100% power.

Subsequently:

- 4D MCC DE-ENERGIZES.

Which one of the following identifies the impact on the Unit 3 Vital DC buses?

A.	Unit 3 Vital DC busses remain un-affected.
B.	ONLY the 3A Vital DC bus loses one Battery Charger
C.	ONLY the 3B Vital DC bus loses one Battery Charger
D.	BOTH the 3A and 3B Vital DC busses lose one Battery Charger each

Proposed Answer: D

A.	Incorrect, plausible since this would be correct for a failure of the D MCC in the same unit.
B.	Incorrect, plausible since this would be correct for the loss of 3C MCC.
C.	Incorrect, plausible since this would be correct for the loss of 3B MCC.

Original for Question # 49.

D.	Correct: 3A2 and 3B2 Battery chargers are both powered from the 4D MCC		
Technical Reference(s)	5610-T-E-1592 SH 1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			NO
Learning Objective:	LP 6902139, Obj. 4		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		8
	55.43		
Components, capacity, and functions of emergency systems.			
Comments:			

Question # 50

Given the following:

- Both units are at 100% reactor power.

Subsequently:

- 4D01, 4B Vital DC Bus, DE-ENERGIZES.

Which one of the following completes the statements below?

Based on the conditions above, a Unit 4 reactor trip __(1)__ occurred.

Unit 4 annunciators are expected to be__(2)__.

A. (1) has
(2) ONLY capable of being half illuminated

B. (1) has
(2) lost

C. (1) has NOT
(2) ONLY capable of being half illuminated

D. (1) has NOT
(2) lost

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. (1) Correct, on loss of 4B DC Buss the 4B Reactor Trip Breaker will lose voltage to UV trip coil, generating a reactor trip.
(2) Correct, on loss of 4B DC Buss the Unit 4 will lose the primary annunciator power supply

B. (1) Correct
(2) Incorrect, Plausible that the candidate will not recognize the procedural requirement on ONOP or the tech spec that applies to both units and remains focused on Unit 4 EOP.

C. (1) Incorrect, Plausible the ONOP for loss of 4B DC bus 4-ONOP-003.4, verifies voltage in the affected bus.
(2) Correct

D. (1) Incorrect, plausible since the loss of other DC buses will NOT generate a Unit 4 reactor trip.
(2) Incorrect, see B(2)

Question Number: 50

Tier: **2** **Group:** 1

K/A: 063 (SF6 ED DC) DC Electrical Distribution
063G2.4.31; Knowledge of annunciator alarms, indications, or response procedures.

Importance Rating: **4.2**

10 CFR Part 55: 41.10

10 CFR 55.43.b : XXX

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References:

Proposed references to be provided: None

Learning Objective: PTN 6900253 Obj. 5

Cognitive Level: Higher X _
Lower _

Question Source: New _
Modified Bank X
Bank _

Question History: Modified question #48 from PTN 2017 exam.

Comments: Screenshot of development references as applicable

1.0 **PURPOSE**

1.1 This procedure provides instructions to be followed in the event DC Buses 4D01 and 4D01A (4B) de-energize, with the unit in Mode 1, Power Operation, and the Auxiliary Transformer is supplying Plant loads. This procedure is to be used after the unit has been stabilized using 4-EOP-ES-0.1, Reactor Trip Response.

2.0 **SYMPTOMS**

2.1 Reactor and Turbine trip

2.2 Power lost to annunciators

2.3 Loss of various Control Room valve position indications

2.4 DC Buses 4D01 and 4D01A (4B) voltmeter indicates 4B DC Bus Voltage is zero.

3.0 **AUTOMATIC ACTIONS**

3.1 Reactor and Turbine trip

3.2 Loss of power to Primary Generator Lockout Relay, which should not prevent automatic Generator trip and automatic fast transfer from Auxiliary to Startup Transformer as backup protection is provided from 4D23.

3.3 Loss of power to 4KV Bus 4A and 4B switchgear synchronism check relays, which prevents manual electrical closure of 4KV Bus 4A and 4B Feeder Breakers from the Auxiliary or Startup Transformers.

3.4 Loss of 4B Bus Load Sequencer.

3.5 Loss of DC power to 4B EDG.

3.6 EDG 4A will remain operable.

3.7 S/G Feedwater Control valves fail closed.

3.8 Train 1 Feedwater Isolation capability of S/G Feedwater Bypass valves.

3.9 Loss of Train 2 AFW.

3.10 Loss of Train B Safeguards equipment.

3.11 4KV Bus 4B breaker control power transfers to DC Bus 3D23.

5.0 SUBSEQUENT ACTIONS

NOTE

- *Turbine Trip capability from Primary Turbine Trip Solenoid, 94/AST, is unavailable.*
- *MSIV closure availability still exists from Train A DC power.*
- *Position indication for MSIVs on the console will be unavailable.*
- *MSIV position will need to be verified locally.*

5.1 Verify DC Buses 4D01 and 4D01A (4B) Voltmeter indicates 4B DC Bus voltage is zero, locally.

5.1.1 **IF** 4B DC Bus Voltmeter indicates 4B DC Buses are Energized, **THEN** inform the Shift Manager, and do **NOT** proceed with this procedure.

5.2 Verify the following breakers at DC Buses 4D01 and 4D01A (4B) Open:

5.2.1 4B Vital Battery Breaker, 4D01-41

5.2.2 Battery Charger 4B1, 4D01-43

5.2.3 Battery Charger 4B2, 4D01-44

5.2.4 Feed from 125VDC Spare Bus D50, 4D01-42

NOTES

- *Loss of 4D01, 4B DC BUS, disables BUS 4B LOAD SEQUENCER and AFW actuation from bus stripping. This requires entry into Technical Specification 3.3.2, Table 3.3-2, Functional Unit 6d, Action 23(a).*
- *Loss of Power signals are lost via the 4B Bus Sequencer, placing the unit in Tech Spec 3.0.3 (Tech Spec 3.3.2, Table 3.3-2, Functional Units 7a, b, and c).*
- *Bus Stripping will NOT automatically occur, the 4B EDG will NOT automatically close in on the bus, and the 4B EDG is Inoperable (actions of Tech Spec 3.8.1.1 apply).*

5.3 Verify that the Inverter (4B or BS), that was supplying 120V Vital Instrument Bus 4P08 before the loss of DC Bus 4D01 **AND** 4D01A (4B), has transferred to CVT 4Y02A.

NOTES

- *The Annunciator System primary power is from 4D01-01 and alternate power is from MCC 4B43, breaker 44335.*
- *The unplanned loss of Safety System Annunciation or Indication in the Control Room may be classifiable under 0-EPIP-20101, Duties of Emergency Coordinator.*

5.4 **IF** annunciators are lost, **THEN** refer to 0-EPIP-20101, Duties of Emergency Coordinator.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	063		2.1.7
	Importance Rating	4.4		

Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question: RO Question # 48

Given the following conditions:

- Unit 4 is at 100% reactor power.

Subsequently:

- 4B DC Bus de-energizes.

Which one of the following completes the statements below?

Based on the conditions above, a reactor trip __(1)__ occurred.

In the event of a subsequent SI signal, the Unit 4 Train B Safeguards equipment __(2)__ AUTOMATICALLY start.

A.	(1) has (2) will NOT
B.	(1) has (2) will
C.	(1) has NOT (2) will NOT
D.	(1) has NOT (2) will

Proposed Answer: A

A.	**Correct** On loss of 4B DC Buss the 4B Reactor Trip Breaker will lose voltage to UV trip coil, generating a reactor trip. At the same time 4B DC Bus supplies power to the 4B sequencer, rendering it unavailable and unable to start 4B train equipment when needed. Incorrect.
B.	**Incorrect** Part 2 incorrect since the 4B sequencer has lost power. Plausible since there is redundancy between Unit when it comes to DC power, i.e. H 480V Load Centers and D 4kV Buses receive redundant control power from opposite unit DC Load Centers.
C.	**Incorrect** Part 1 incorrect since there is a reactor trip takes precedence over the loss of power investigation. Plausible since the ONOP for loss of 4B DC bus 4-ONOP-003.4, verifies voltage in the affected bus.
D.	**Incorrect** Part 1 incorrect since there is a reactor trip takes precedence over the loss of power investigation. Plausible since the ONOP for loss of 4B DC bus 4-ONOP-003.4, verifies voltage in the affected bus. Part 2 incorrect since the 4B sequencer has lost power. Plausible since there is redundancy between Unit when it comes to DC power, i.e. H 480V Load Centers and D 4kV Buses receive redundant control power from opposite unit DC Load Centers.

Technical Reference(s)	4-ONOP-003.4		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902253 Obj. 1		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		

Question # 51

Which one of the following completes the statements below regarding EDGs?

For Unit 3, a Generator Phase Bus Differential condition __(1)__ prevent a NON-EMERGENCY start.

EDG Lockout Reset Buttons __(2)__ present in the Control Room.

A. (1) will
   (2) are NOT

B. (1) will
   (2) are

C. (1) will NOT
   (2) are NOT

D. (1) will NOT
   (2) are

Answer Analysis

A. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since there are EDG controls for resetting alarms that are just locally at the EDG. Also the EDG lockout relay is located in the EDG building.

B. Correct. The EDG Overcurrent Phase trip will lockout each unit's EDGs. The EDG Lockout is provided with the relay handle locally and with reset pushbuttons in the control room.

C. Incorrect.
Part 1 incorrect, plausible since it is a common misconception that non-emergency trips are the same as non-emergency starts.
Part 2 incorrect, plausible since there are EDG controls for resetting alarms that are just locally at the EDG. Also the EDG lockout relay is located in the EDG building.

D. Incorrect.
Part 1 incorrect, plausible since it is a common misconception that non-emergency trips are the same as non-emergency starts.
Part 2 correct.

Question Number: 51

Tier: **2** **Group:** 1

K/A: **064AA3.06;** Ability to monitor automatic operation of the EDG system, including: Start and stop

Importance Rating: **3.3**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, have knowledge of plant differences, and identify automatic EDG system responses.

SRO Justification: N/A

Technical References: 5613-T-L1 SH-9A5

Proposed references to be provided: None

Learning Objective: PTN 6902136 OBJ. 7

Cognitive Level: Higher

Lower X_

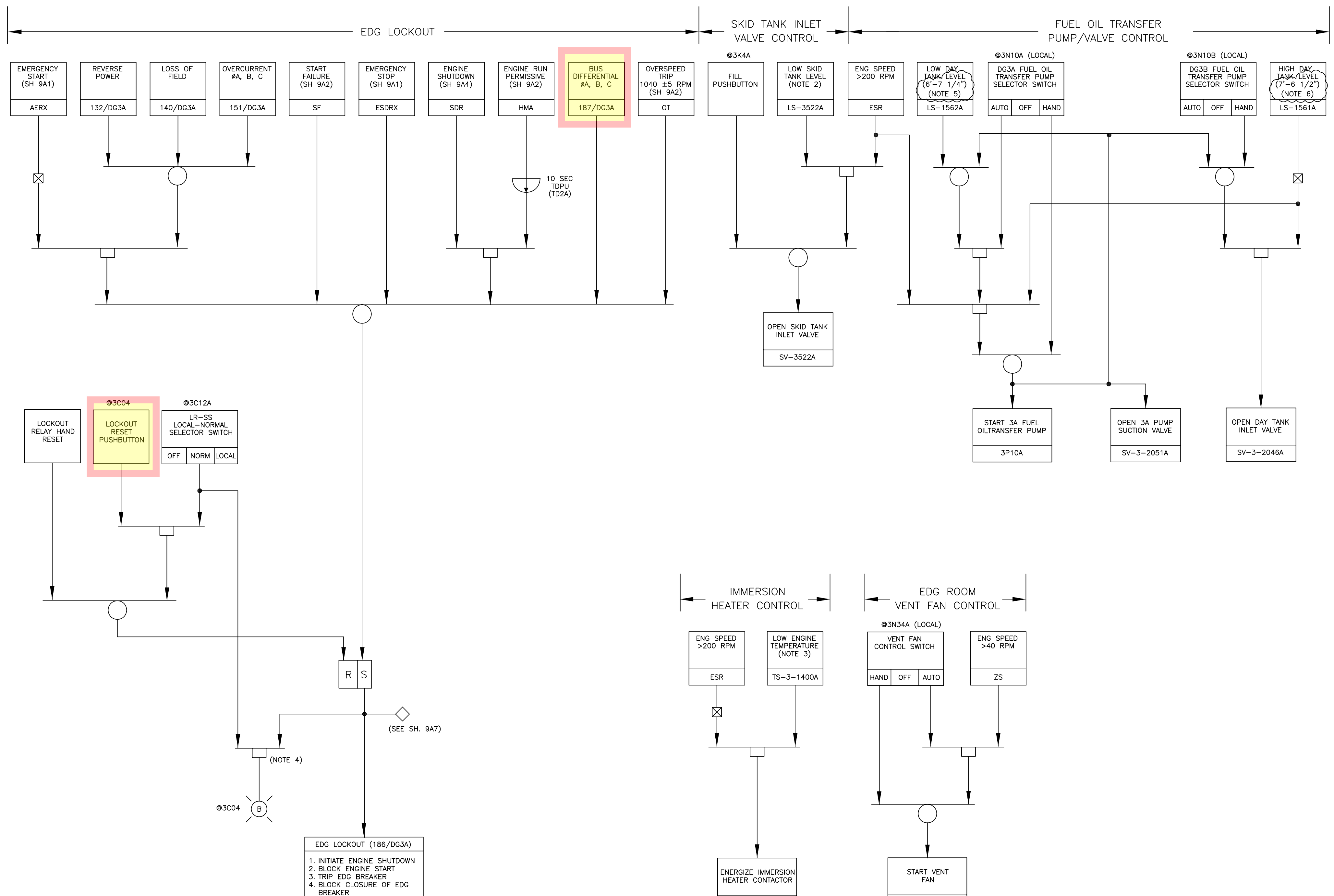
Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable



NOTES:

1. LOGIC SHOWN FOR DG3A. LOGIC FOR DG3B IS IDENTICAL WITH APPROPRIATE A TO B CHANGES TO COMPONENT/PANEL NUMBERS. REFER TO SCHEMATIC DIAGRAMS FOR CHANGES.
2. SKID TANK LEVEL SWITCH IS SET TO OPEN THE SOLENOID VALVE 15" FROM TOP OF TANK AND CLOSE IT 6" FROM TOP OF TANK.
3. IMMERSION HEATER TEMPERATURE SWITCH IS SET TO ENERGIZE THE IMMERSION HEATER AT 125° F AND DE-ENERGIZE IT AT 155° F.
4. LOCKOUT RELAY BLUE LIGHT GLOWS STEADY WHEN RESET AND FLASHES WHEN ACTUATED. FOR ANY EDG LOCKOUT, INSPECT ALL RELAYS ON GENERATOR PANEL AND 4KV BREAKER FOR DROPPED TARGET FLAGS. DOCUMENT AND ADVISE ELECTRICAL DEPARTMENT. RESTART ONLY AFTER CAUSE HAS BEEN FOUND AND CORRECTED.
5. 3B EDG LS-1562B LEVEL SET POINT IS (6'-8").
6. 3B EDG LS-1561B LEVEL SET POINT IS (7'-7").

NUCLEAR SAFETY RELATED

POD

NO	DATE	REVISION	BY	CH	COR	APP
2	10-8-08	ISSUED AS-BUILT PER PC/M 08-151.	RH	RV	–	JAF
1	2-15-06	ISSUED AS-BUILT PER CRN-E-17083(PC/M 03-106).	RV	RH	–	TS
0	7-7-92	AS-BUILT PER DCR TPI-91-419 REV. 2	RH	ERW	ET	FV

FPL

BY: S HEWITT
CH
COR
APP
DATE

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT NO 3
LOGIC DIAGRAM
EDG LOCKOUT AND
ENGINE AUXILIARIES

DRAWING NUMBER 5613-T-L1 SH 9A5

DISC I&C

REV 2

5613-T-L1_SH 9A5.DWG

Question # 52

- Unit 3 has Refueling activities ongoing in the containment.

Subsequently:

- The HIGH alarm for R-3-11, Containment Particulate PRMS Channel, actuates.

Which one of the following completes the statements below?

The Refueling Manipulator Crane __(1)__ AUTOMATICALLY stop upward hoist movement due to the R-3-11 alarm.
The Control Room Ventilation System __(2)__ receive a signal to align for recirc mode.

A. (1) will
(2) will NOT

B. (1) will NOT
(2) will

C. (1) will NOT
(2) will NOT

D. (1) will
(2) will

Answer Analysis

Discussion: Correct answer is B R-11 and R-12 are the Containment Process Rad Monitors**.** If they alarm both containment isolation and control room ventilation isolation automatically occur. The Refueling Manipulator crane does have interlocks that will prevent upward movement but R-11 and / or R-12 in alarm trip is not one of them

A1 – Incorrect. Plausible because R-11 and R-12 are in containment and the Manipulator hoist does have interlocks that will stop upward movement.
A2 is Incorrect. Plausible Same as C2

B – Correct WILL NOT and WILL

C1 – Correct
C2 – Incorrect. Plausible because the control room is not part of containment and it is not intuitive that a containment rad alarm should isolate control room ventilation. Containment Area alarms will not isolate containment. PER the ARP H 5/2 Unit 4 may be slightly different and also have control room recirculation alignment occur.

D1 Plausible- Same as A1
D2 Correct

Question Number: 52

Tier: **2** **Group:** 1

K/A: 073 (SF7 PRM) Process Radiation Monitoring
073AK5.02; Knowledge of the operational implications as they apply to concepts as they apply to the PRM system:
Radiation theory, including sources, types, units, and effects

Importance Rating: **2.5**

10 CFR Part 55: 41.5

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 5610-T-L1 SH11

Proposed references to be provided: None

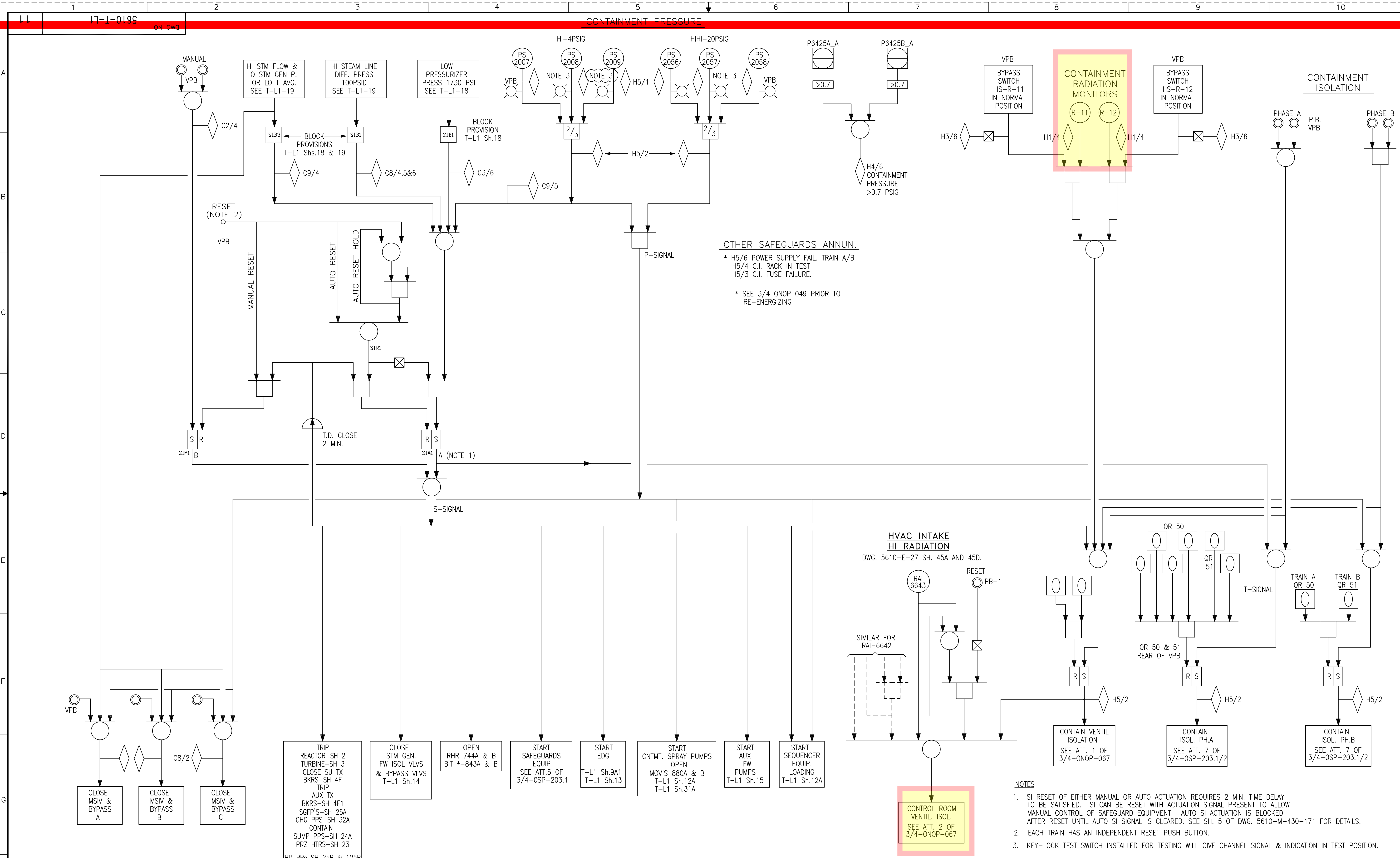
Learning Objective: PTN 6902163 OBJ. 8

Cognitive Level: Higher _
Lower X_

Question Source: New X_
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable



NOTES

1. SI RESET OF EITHER MANUAL OR AUTO ACTUATION REQUIRES 2 MIN. TIME DELAY TO BE SATISFIED. SI CAN BE RESET WITH ACTUATION SIGNAL PRESENT TO ALLOW MANUAL CONTROL OF SAFEGUARD EQUIPMENT. AUTO SI ACTUATION IS BLOCKED AFTER RESET UNTIL AUTO SI SIGNAL IS CLEARED. SEE SH. 5 OF DWG. 5610-M-430-171 FOR DETAILS.
2. EACH TRAIN HAS AN INDEPENDENT RESET PUSH BUTTON.
3. KEY-LOCK TEST SWITCH INSTALLED FOR TESTING WILL GIVE CHANNEL SIGNAL & INDICATION IN TEST POSITION.

SAFETY RELATED | POD

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
33	10-19-17	ISSUED AS-BUILT PER EC 289667.	RH	CW	–	BB	27	07-15-03	ISSUED AS-BUILT PER CRN-E-16419 (PC/M 02-065).	RV	RH	BP	TS
32	2-22-16	ISSUED AS-BUILT PER EC-DCR 285837.	RH	AFG	–	SB	26	1-20-98	ISSUED AS-BUILT PER CRN'S I-3659 & I-3663(PC/M 97-042).	RV	RH	MA	TS
31	08-28-12	ISSUED AS-BUILT PER EC 246935 (PC/M 08-187).	RV	RH	–	PMB	25	11-7-97	ISSUED AS-BUILT PER CRN-I-3640(PC/M 97-036)	RV	RH	MA	TS/JG
30	12-28-09	ISSUED AS-BUILT PER CRN-E-18083 (PC/M 09-106).	RV	BB	–	LG	24	1-23-97	ISSUED AS-BUILT PER CRN-I-3482(PC/M 96-080)	RV	RH	BP	TS
29	11-20-09	ISSUED AS-BUILT PER PC/M 08-037.	RV	RH	–	AD	23	9-30-96	INCORPORATED CRN-I-3418 FOR PC/M 96-058.	RV	BP	–	TS
28	4-9-09	ISSUED AS-BUILT PER PC/M 08-006.	RH	KCP	–	RCS	34	–	ISSUED AS-BUILT PER EC 291574.	SEE DUP FOR INITIALS & DATE			

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

FLORIDA POWER & LIGHT

SAFEGUARDS ACTUATION AND STEAM LINE ISOLATION

POD

DRAWING NUMBER 5610-T-L1

SYS 063

REV 34

SHEET 11

5610-T-L1_SH 11.DWG

24 x 36 D SIZE

Question # 53

Which one of the following completes the statements below?

In order to ensure that emergency heat loads have adequate cooling, the ICW System is designed to AUTOMATICALLY __(1)__ when __(2)__.

A. (1) isolate ICW from TPCW
(2) CCW temperature is too high

B. (1) isolate ICW from TPCW
(2) Safety Injection is actuated

C. (1) align ICW supplemental cooling to ICW
(2) CCW temperature is too high

D. (1) align ICW supplemental cooling to ICW
(2) Safety Injection is actuated

Answer Analysis

Discussion: The only auto feature regarding ICW heat loads is isolation of TPCW on safety injection.

A. (1) Correct
(2) Incorrect: Plausible because this would divert more flow to the CCW heat exchangers, but does not automatically happen on high temperature.

B. Correct, see discussion above

C. (1) Incorrect Plausible because this would help with removing heat from CCW but it does not happen automatically.
(2) Incorrect, see D(2)

D. (1) Incorrect, see C(1)
(2) Incorrect plausible because ICW supplemental cooling would help control CCW temperature, but it does not automatically align on any signal.

Question Number: 53

Tier: **2** **Group:** 1

K/A: **076AA3.02**
Ability to monitor automatic operation of the SWS, including: Emergency heat loads

Importance Rating: **3.7**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** N/A

K/A Match: The K/A is matched as the applicant must understand automatic ISW actions.

SRO Justification: N/A

Technical References: 5613-M-3019 SH 1

Proposed references to be provided: None

Learning Objective: PTN 6900154, OBJ. 6

Cognitive Level: Higher
Lower X_

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

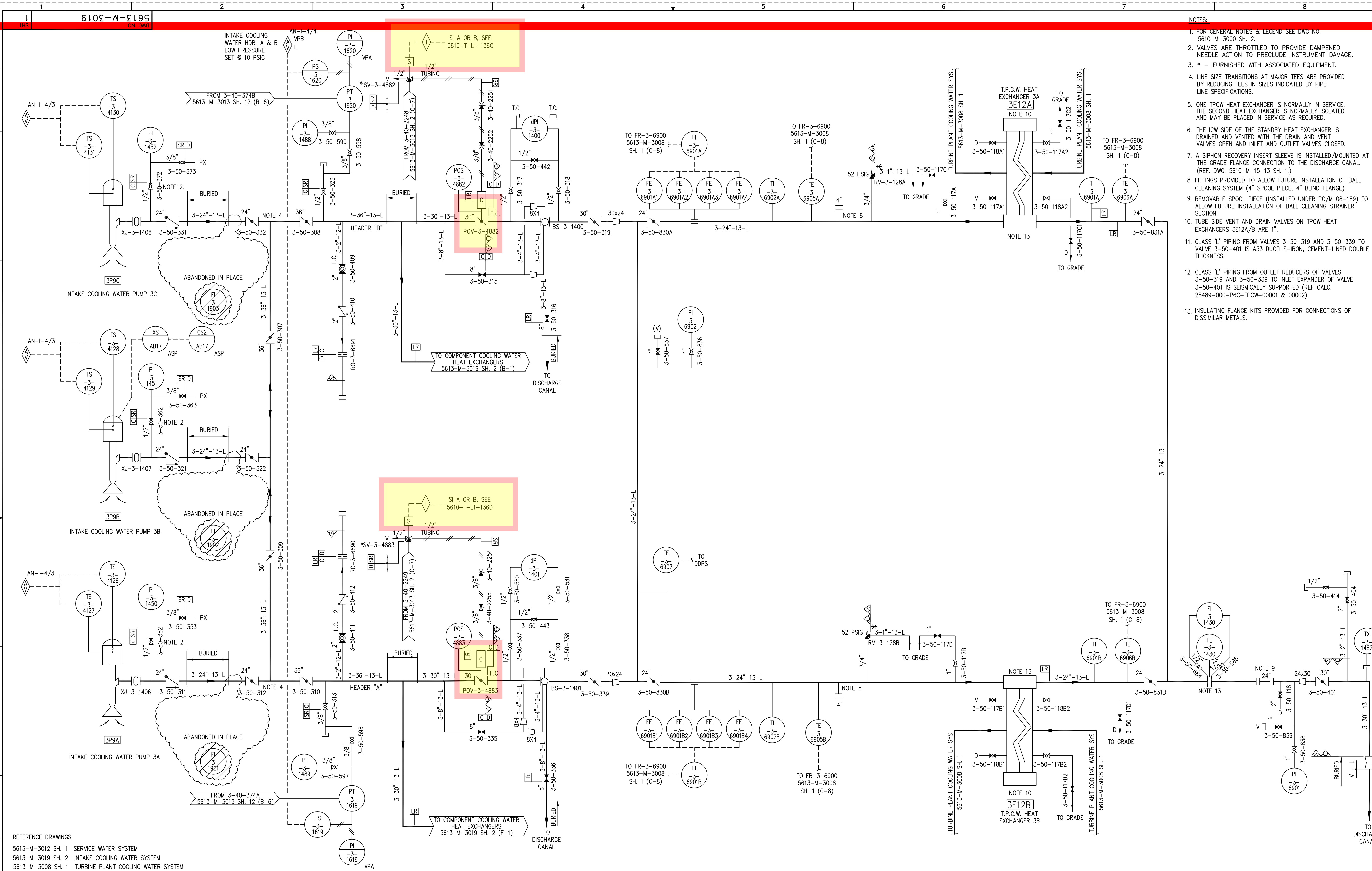
1.1. ICW/TPCW Isolation Valves

The ICW-to-TPCW isolation valves (POV-4882 and POV-4883) are air-to-open and fail closed, on loss of air or electrical power. **A dedicated air accumulator is provided to hold these valves open for a minimum of two hours, in the event that the instrument-air source is lost, to preclude damage to turbine-related equipment**. The valves may also be opened manually, via adjacent handwheel actuators (note that the handwheels have been disconnected from the actuator stems, to maintain the seismic qualification of the valves, and are staged locally).

Three-position OPEN-AUTO-CLOSE (spring return to AUTO) control switches, along with associated valve-position-indicating lights, are provided in the control room; local control is provided by two-position REMOTE-CLOSE switches. If the valves have been opened from the control room, the local control stations may only be used to close them; however, if the control room (or a safety-injection signal) has closed the valves, the local stations have no operating ability. The control room can reposition the valves only when the local control switches are in the REMOTE position.

The valve-control circuits are designed such that a safety-injection actuation will override all control stations and automatically close the valves, thereby allowing ICW to supply only the CCW loads. The valve-closure signal from a safety-injection actuation can be subsequently reset, by placing the control room switch in the OPEN position (after the safety-injection signal has been reset).

The supplemental cooling system lowers the water temperature in either the Unit 3 or Unit 4 ICW system to reduce the temperature of the CCW system and subsequently lower the Containment temperature. It has no auto functions.



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2.
2. VALVES ARE THROTTLED TO PROVIDE DAMPENED NEEDLE ACTION TO PRECLUDE INSTRUMENT DAMAGE.
3. * – FURNISHED WITH ASSOCIATED EQUIPMENT.
4. LINE SIZE TRANSITIONS AT MAJOR TEES ARE PROVIDED BY REDUCING TEES IN SIZES INDICATED BY PIPE LINE SPECIFICATIONS.
5. ONE TPCW HEAT EXCHANGER IS NORMALLY IN SERVICE. THE SECOND HEAT EXCHANGER IS NORMALLY ISOLATED AND MAY BE PLACED IN SERVICE AS REQUIRED.
6. THE ICW SIDE OF THE STANDBY HEAT EXCHANGER IS DRAINED AND VENTED WITH THE DRAIN AND VENT VALVES OPEN AND INLET AND OUTLET VALVES CLOSED.
7. A SIPHON RECOVERY INSERT SLEEVE IS INSTALLED/MOUNTED AT THE GRADE FLANGE CONNECTION TO THE DISCHARGE CANAL. (REF. DWG. 5610-M-15-13 SH. 1.)
8. FITTINGS PROVIDED TO ALLOW FUTURE INSTALLATION OF BALL CLEANING SYSTEM (4" SPOOL PIECE, 4" BLIND FLANGE).
9. REMOVABLE SPOOL PIECE (INSTALLED UNDER PC/M 08-189) TO ALLOW FUTURE INSTALLATION OF BALL CLEANING STRAINER SECTION.
10. TUBE SIDE VENT AND DRAIN VALVES ON TPCW HEAT EXCHANGERS 3E12A/B ARE 1".
11. CLASS 'L' PIPING FROM VALVES 3-50-319 AND 3-50-339 TO VALVE 3-50-401 IS A53 DUCTILE-IRON, CEMENT-LINED DOUBLE THICKNESS.
12. CLASS 'L' PIPING FROM OUTLET REDUCERS OF VALVES 3-50-319 AND 3-50-339 TO INLET EXPANDER OF VALVE 3-50-401 IS SEISMICALLY SUPPORTED (REF CALC. 25489-000-P6C-TPCW-00001 & 00002).
13. INSULATING FLANGE KITS PROVIDED FOR CONNECTIONS OF DISSIMILAR METALS.

REFERENCE DRAWINGS

5613-M-3012 SH. 1 SERVICE WATER SYSTEM
5613-M-3019 SH. 2 INTAKE COOLING WATER SYSTEM
5613-M-3008 SH. 1 TURBINE PLANT COOLING WATER SYSTEM
5613-M-3030 SH. 1 COMPONENT COOLING WATER SYSTEM
5610-T-L1 SH. 136C TPCW/ICW ISOLATION VALVE POV-3-4882 CHANNEL A.
5610-T-L1 SH. 136D TPCW/ICW ISOLATION VALVE POV-3-4883 CHANNEL B.

SPEC MN-3.11 PIPING CLASS SHEETS

THIS DRAWING SUPERSEDES DRAWINGS:		
5610-M-5	SH. 1	REV. 41
5610-M-342	SH. 1	REV. 9
5610-M-342	SH. 6	REV. 0
5610-M-364		REV. 2
5610-M-365	SH. 1	REV. 5

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4064 SH. 13 REV. 1
FPL POD 5610-T-E-4065 SH. 1 REV. 39
FPL POD 5610-T-E-4065 SH. 2 REV. 58

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
35	01-06-12	ISSUED AS-BUILT PER EC 242515 (PC/M 09-089).	RH	RV	–	LG	39	9-11-13	ISSUED AS-BUILT PER CRN-222 (EC 246937) ITOP 13-09-006.	RH	BB	–	PSB
34	9-24-09	ISSUED AS-BUILT PER CRN M-12844 (PC/M 09-070).	RH	RV	BSG	PRB	38	8-15-13	ISSUED AS-BUILT PER CRN-220 (EC 246937) ITOP 13-08-024.	RH	RV	CP	JD
33	2-20-09	ISSUED AS-BUILT PER PC/M 08-028.	RH	RV	BSG	SB	37	12-15-12	ISSUED AS-BUILT PER CRN-182 (EC 246937). ITOP 12-08-382	RV	BB	–	PSB
32	4/7/06	ISSUED AS-BUILT PER PC/M 05-094.	RH	JPG	PRB	JTL	36	7-3-12	ISSUED AS-BUILT PER EC 246937 (PC/M 08-189).	RH	RV	JTB	–
41	–	ISSUED AS-BUILT PER EC 290896.	SEE DUP FOR SIGNATURES				0	1-29-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	MAB	MD	SVM/RAS	LRB
40	4-18-14	ISSUED AS-BUILT PER 278832 (278832-FINAL).	RV	BB	–	CC	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-90-541.	—	—	—	—

POD

FPL

TURKEY POINT NUCLEAR UNIT 3

P & I D

INTAKE COOLING WATER SYSTEM

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3019 | SYS 019

SHEET 1 | REV 41

5613-M-3019_SH 1

Question # 54

Given the following:

- Both Units are at 100% power.

Subsequently:

- Instrument Air Pressure on Unit 3 begins to lower.
- All Instrument Air Compressors are OFF and will NOT start.

Which one of the following completes the statements below?

3-ONOP-013, Loss of Instrument Air, directs that the Instrument Air System be supplied by __(1)__.
The Instrument Air System may be restored to the NORMAL lineup when a MINIMUM of __(2)__ Instrument Air Compressor(s) ~~are~~ running.
is/are

A. (1) connecting a Temporary Diesel Instrument Air Compressors
(2) one

B. (1) connecting a Temporary Diesel Instrument Air Compressors
(2) two

C. (1) cross tying to the Service Air System
(2) one

D. (1) cross tying to the Service Air System
(2) two

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. (1) Incorrect, Plausible because there are Temporary Diesel Instrument Air Compressors that can be used during Maintenance Periods when there are two IAC's out of service. The NOP discusses this use in the prerequisites, and there is a separate NOP for Temp Diesel IAC operation.
(2) Correct, Only one IAC is required to be restored to return to IAC supply of Instrument Air.

B. (1) Incorrect, see A(1)
(2) Incorrect, Plausible since both Service Air Compressors must be running or the procedure provides action to reduce Service Air Loads when crosstied with Instrument Air.

C. C. Correct, IAW 0-NOP-013 and 3-ONOP-013.

D. (1) Correct
(2) Incorrect, see B(2)

Question Number: 54

Tier: 2 **Group:** 1

K/A: 078 (SF8 IAS) Instrument Air
078AK1.04 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems:
Service air

Importance Rating: **2.7**

10 CFR Part 55: 41.2 to 41.9

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-ONOP-013, 0-NOP-013

Proposed references to be provided: None

Learning Objective: PTN 6902286 OBJ. 7 & 8

Cognitive Level: Higher _

Lower _X

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

5.0 INFREQUENT OPERATIONS

5.1 Service Air System Tie to Unit 3 IA

NOTE

This section is intended for use upon loss of all Instrument Air Compressors. Restoration from this lineup should be performed upon start of any Instrument Air Compressor.

1. **ENSURE** both Service Air Compressors in operation per 0-NOP-013.01, Service Air System.

2. **ENSURE** PI-3-1492, SERV AIR PRESS on VPA is greater than 75 PSIG.

3. **OPEN** 40-2059, SERVICE AIR SUPPLY TO UNIT 3 / UNIT 4 TIE VALVE on mezzanine SW of 3A Heater Drain Tank.

4. **ENSURE** PI-3-1444, INST AIR PRESS on VPA stabilizes at or above 75 PSIG.

5. IF PI-3-1444, INST AIR PRESS continues to lower, THEN **REFER TO** 3-ONOP-013, Loss of Instrument Air.

6. WHEN at least one Instrument Air Compressor is operating, THEN:

   A. While monitoring PI-3-1444, INST AIR PRESS, slowly **CLOSE** 40-2059, SERVICE AIR SUPPLY TO UNIT 3 / UNIT 4 TIE VALVE.

   B. **CHECK** PI-3-1444, INST AIR PRESS as least 95 PSIG.

End of Section 5.1

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.2 Subsequent Operator Actions (continued)

NOTE

40-2059, SERVICE AIR BACKUP TO UNITS 3/4 INSTRUMENT AIR ISOLATION VLV, is located on mezzanine SW of 3A Heater Drain Tank.

2. **START** all available Instrument Air Compressors and **DISPATCH** personnel to determine cause of lowering Instrument Air pressure.

   A. **REFER TO** Attachment 1, Manual Start of Instrument Air Compressors.

   Response Not Obtained: IF Unit 3 Instrument Air Compressors can **NOT**, THEN **PERFORM** Attachment 5:

3. **CHECK** Instrument Air Compressor(s) as suspected source of air loss:

   A. IF any Unit 3 Instrument Air Compressor is confirmed as the source of air loss, THEN **CLOSE** the applicable compressors discharge valve:

      * IAS-3-058, 3CM COMPRESSOR OUTLET ISOLATION VALVE
      * IAS-3-059, 3CD COMPRESSOR OUTLET ISOLATION VALVE

ATTACHMENT 5
Responding To A Total Loss Of Instrument Air
(Page 1 of 4)

1.0 RESPONDING TO A TOTAL LOSS OF INSTRUMENT AIR

1. **NOTIFY** maintenance to RESTORE an Instrument Air compressor to service.

> **NOTE**
>
> 40-2059, SERVICE AIR BACKUP TO UNITS 3/4 INSTRUMENT AIR ISOLATION VALVE, is located on the mezzanine deck SW of 3A Heater Drain Tank.

2. **DETERMINE** if Service Air is available as a source of instrument air by performing the following:

   A. **ENSURE** Service Air system is in service per 0-NOP-013.01, Service Air System.

   B. **ENSURE** all available Service Air compressors are operating.

   C. At Shift Managers discretion, **PERFORM** the following:

      (1) **OPEN** 40-2059, SERVICE AIR BACKUP TO UNITS 3/4 INSTRUMENT AIR ISOLATION VLV.

      (2) **ENSURE** OPEN 3-40-408, INSTRUMENT AIR CROSSTIE HEADER UNIT 3 ISOLATION VALVE.

   D. IF only one Service Air compressor is available, THEN **SECURE** non-required service air loads.

3. **DETERMINE** if the opposite unit Instrument Air system is available as a source of instrument air by PERFORMING the following:

   A. IF there is sufficient air pressure from opposite unit compressor(s) AND CV-3-1605, UNIT 3 INSTRUMENT AIR CROSSTIE ISOLATION CONTROL VALVE, is closed or isolated due to low air pressure on Unit 3 (less than 80 psig), THEN **ESTABLISH** Instrument Air to Unit 3 as follows:

      (1) **VERIFY** OPEN 4-40-408, INSTRUMENT AIR CROSSTIE HEADER UNIT 4 ISOLATION VALVE.

Question # 55

Given the following:

- Unit 3 RCS is at 100% power.

Which one of the following completes the statements below?

In order to prevent pressure build up in the containment, __(1)__ Containment Instrument Air Bleed Isolation Valve(s) is(are) __(2)__.

A. (1) ONLY 1
(2) maintained OPEN

B. (1) ONLY 1
(2) cycled as required and restored to CLOSED

C. (1) BOTH
(2) maintained OPEN

D. (1) BOTH
(2) cycled as required and restored to CLOSED

Answer Analysis

A. Incorrect.
Part 1 incorrect, plausible since operating a single IA bleed valve will lower containment pressure.
Part 2 correct.

B. Incorrect.
Part 1 incorrect, plausible since operating a single IA bleed valve will lower containment pressure.
Part 2 incorrect, plausible since cycling the IA bleed valves would also control containment pressure.

C. Correct. Normally both IA bleed valves are maintained open in order to limit containment pressure.

D. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since cycling the IA bleed valves would also control containment pressure.

Question Number: 55

Tier: 2 **Group:** 1

K/A: **103AA1.01 Ability to predict and/or monitor changes in parameters** (Containment pressure, temperature, and humidity) **to prevent exceeding design limits associated with operating the containment system controls:**

Importance Rating: **3.7**

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: **RO Question addressing overall Mitigating strategy with Purge valve unavailable.** The K/A is matched as the applicant must evaluate plant conditions, identify the proper containment pressure control system and operator action required to mitigate the event.

SRO Justification: N/A

Technical References: 3-NOP-053

Proposed references to be provided: None

Learning Objective: PTN 6902129 OBJ. 9

Cognitive Level: Higher

Lower _X

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

ALARM	AUTO FUNCTION	SENSING DEVICE	SETPOINT	ANNUNCIATOR
CONTAINMENT HIGH PRESSURE SI	Phase A isolation, RX trip, S.I.	3(4)-PS-2008 3(4)-PS-2007 3(4)-PS-2009	2:4@ 4 PSIG ↑	C-9/5
CONT. HI-HI PRESS.	- -	3(4)-PS-2057 3(4)-PS-2056 3(4)-PS-2058	20 PSIG ↑	VPB status light
CONTAINMENT HIGH AND HI-HI PRESSURE	Phase A & B isol., S.I., RX trip, Hi & Hi-Hi Cont. Spray	3(4)-PS-2007, 2008, & 2009 3(4)-PS-2056, 2057 & 2058	1:3 @ 4 PSI ↑ 1:3 @ 20 PSI ↑	H-5/1

1.0 PURPOSE

This procedure provides the instructional guidance for operation of the Unit 3 Containment Purge System.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

1. A Radioactive Gas Release Permit is **NOT** required for CV-3-2819 and CV-3-2826, CNTMT INSTRUMENT AIR BLEED ISOL valves. These valves shall be normally open to prevent buildup of pressure in the containment.

2. If 3-12-031, SFP TRANSFER CANAL GATE VALVE, is open, the refueling cavity is filled with water, and refueling integrity is established; then a slow increase in containment pressure will occur and cause the SFP level to increase when the Containment Purge System is secured. Use of air operated equipment inside containment will result in a more rapid increase in containment pressure and will require monitoring SFP level more closely. (OE 3989)

3. The Containment Purge Exhaust Fan from either unit may be used for purging operation. If 4V20, U-4 CNTMT PURGE EXHAUST FAN, from the opposite unit is used, it will **NOT** automatically trip from any of the following conditions:

   - High containment activity on R-3-11, PARTICULATE Containment Radiation Monitor
   - High containment gaseous activity on R-3-12, GASEOUS Containment Radiation Monitor
   - Automatic or manual Safety Injection signal
   - Containment Isolation Phase A signal
   - Containment Isolation Phase B signal

Question # 56

Given the following:

- Unit 3 is in MODE 3.
- 3-GOP-301, Hot Standby to Power Operation, is in progress.

Subsequently:

- Reactor Trip Breakers have been reset

Which one of the following completes the statements below?

In order to reset the Group Step Counters the crew will depress the __(1)__ pushbutton. IAW 3-GOP-301, before withdrawing any rod bank from the fully inserted position, the Analog Rod Position Indicators must be within a MAXIMUM of __(2)__ steps of the Group Step Counters.

A. (1) Rod Control Alarm Reset
   (2) 18 steps

B. (1) Rod Control Alarm Reset
   (2) 12 steps

C. (1) Rod Control Startup Reset
   (2) 18 steps

D. (1) Rod Control Startup Reset
   (2) 12 steps

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. (1) Incorrect, Plausible because this is a required step of GOP-301, but this does not reset the Bank Rod Indication to zero. Possibility the candidate does not understand the reset function.
(2) Correct, as per 3-OSP-201.1, RO Daily Logs, Attachment 2 for Mode 2 surveillances, which must be completed prior to reactor startup.

B. (1) Incorrect, See A(1)
(2) Incorrect, Plausible since this is the requirement for rod alignment >90% power. The candidate may fail to the more conservative number if the power requirement is not known.

C. (1) Correct, as per GOP-301
(2) Correct

D. (1) Correct
(2) Incorrect, see B(2)

Question Number: 56

Tier: 2 **Group:** 2

K/A: 014 (SF1 RPI) Rod Position Indication
A4.04; Ability to manually operate and/or monitor in the control room:
Re-zeroing of rod position prior to startup

Importance Rating: 2.7

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-OSP-201.1, 3-GOP-301, 5610-T-D-12A

Proposed references to be provided: None

Learning Objective: PTN 6902407 OBJ. 5

Cognitive Level: Higher X _
Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

4.0 PRECAUTIONS/LIMITATIONS

4.1 Criticality should be anticipated anytime when shutdown or control rod banks are being withdrawn or boron dilution is in progress.

4.2 All shutdown rods shall be fully withdrawn before the reactor is made critical.

4.3 Do not make the reactor critical with a moderator temperature coefficient of reactivity more positive than +5 pcm/°F (except as permitted for low power physics tests).

4.4 The approach to criticality shall be guided by plotting inverse count rate ratio versus control rod position. Observe the 1/m plot to assure criticality will not occur below the insertion limit for zero power.

4.5 Before withdrawing any rod bank from the fully inserted position, the group step counters and the rod position indicators for that bank shall meet the control rod position Acceptance Criteria in 3-OSP-201.1, RO Daily Logs.

4.6 When moving shutdown or control rod banks; the Group Step Counters, RPIs, and all Nuclear Instrumentation Channels shall be closely monitored to verify proper bank movement and bank overlap for control rods.

4.7 The Reactor Coolant System lowest operating loop temperature (Tavg) shall be greater than or equal to 541°F with Keff greater than or equal to 1.0.

4.8 All Reactor coolant loops shall be in operation prior to making the reactor critical, Mode 2. With less than 3 Loops in operation, restore all Loops to operable status or be in Hot Standby within six (6) hours.

4.9 Before transferring the Rod Control selector from Manual to Auto mode, the control rod banks shall be positioned as required to adjust Tavg within 1.0°F of Tref.

4.10 At power, all Rod Position Indicators and Power Range Nuclear Channels shall be periodically monitored for control rod misalignment and abnormal power distribution.

4.11 Every attempt should be made to maintain the Axial Flux Difference within the Operational Space to avoid; otherwise, unnecessary power reductions; reference 0–NOP–059.09, Operation Within the Axial Flux Difference Operational Space.

4.12 Control banks shall be maintained above the respective Rod Bank A-B-C or D Low Limit Alarm by maintaining the required RCS boron concentration.

4.13 When any control rod bank is below the Rod Bank A-B-C or D Extra Low Limit Alarm, then refer to T.S. 3.1.3.6, Control Rod Insertion Limits.

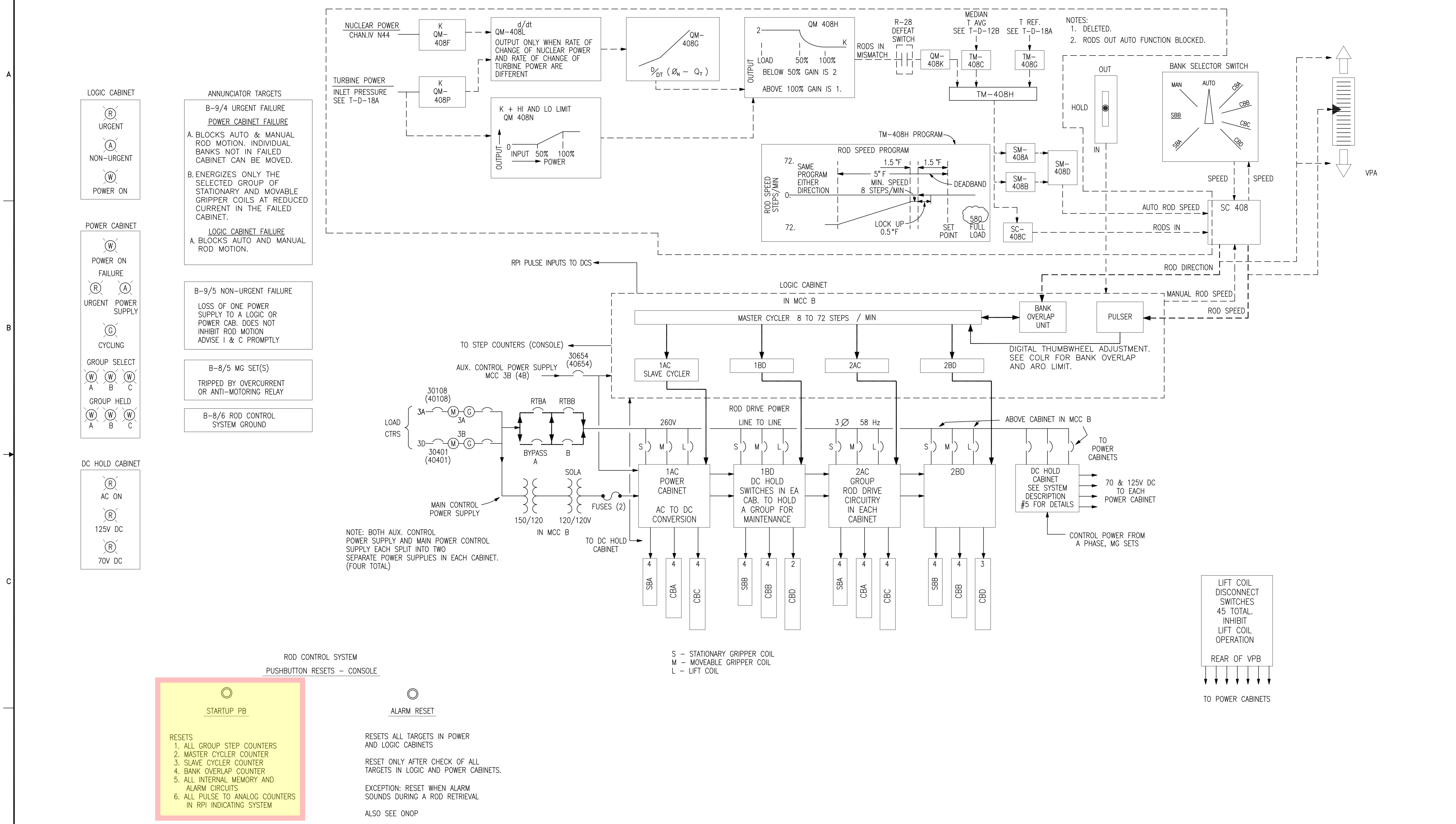
4.14 SUR should not be permitted to exceed a steady state value of 1.0 dpm below the POAH and 0.5 dpm above the POAH.

4.15 If the Steam Dump System is automatically armed by a load rejection and equilibrium conditions are re-established, the Steam Dump Control shall be reset by placing the steam dump to condenser Mode Selector switch to Reset.

ATTACHMENT 1
(Page 6 of 16)

MINIMUM INSTRUMENTATION AND EQUIPMENT LIST - MODE 1

TECH SPEC MINIMUM OPERABLE EQUIPMENT LIST				
EQUIPMENT	**MODE**	**1**		
	SHIFT	**M**	**D**	**P**
1. Boron Injection Flowpath a. BAST - BA pump - charging pump suction **AND** b. 1/2 paths RWST - charging pump suction **AND** c. Charging pump discharge - Regen HX – RCS	3.1.2.2			
2. Borated Water Sources a. BA Storage System i. Borated water volume and concentration IAW T.S. 3.1.2.5 ii. B Tank room >62°F b. RWST i. >320,000 gallons ii. 2400 to 2600 ppm iii. >39°- 100°F	3.1.2.5			
3. 2/3 Charging pumps operable	3.1.2.3			
4. a. All full-length shutdown & control rods Analog RPI w/i ± 12 steps of group demand counter (N/A if Pwr ≤ 90%) b. All full-length shutdown & control rods Analog RPI w/i ± 18 steps of group demand counter (N/A if Pwr > 90%)	3.1.3.1			
5. a. Bank Analog RPIs ± 12 steps (N/A if Pwr ≤ 90%) b. Bank Analog RPIs ± 18 steps (N/A if Pwr > 90%) c. Group demand counters ± 2 steps	3.1.3.2			
6. 3/3 Reactor Coolant Loops in operation	3.4.1.1			
7. 3/3 Pressurizer Code Safeties	3.4.2.2			
8 Pressurizer with ≤92% level and 2/2 Groups Pressurizer BU Heaters w/125 KW per group capable of being supplied by emergency power	3.4.3			
9. 2/2 Pressurizer PORV Block Valves	3.4.4			
10. 3/3 Steam Generators	3.4.5			
11. RCS Leak Detection System a. R-11 **OR** R-12 **AND** b. Containment Sump Level	3.4.6			
12. RCS vent paths of 2 valves in series a. Rx Vessel head **AND** b. Pressurizer Steam Space	3.4.11			



REV	DATE	REVISION	BY	CH	APP	APP
16	1-20-13	ISSUED AS-BUILT PER CRN-031 (EC 247048) (ITOP 13-01-028)	AFG	BB	–	JAD
15	9/19/11	ISSUED AS-BUILT PER EC-DCR 273667.	RV	RH	–	LG
14	12/17/07	ISSUED AS-BUILT PER PC/M 07-019.	RV	BB	–	RJT
13	6-2-06	ISSUED AS-BUILT PER CRN-I-4427 (PC/M 03-106).	RV	RH	–	TS
12	5-7-04	ISSUED AS-BUILT PER CRN-E-16603 (PC/M 03-106).	RV	RH	–	PDS
11	2-12-04	ISSUED AS-BUILT PER CRN-I-4131 (PC/M 03-050).	RV	RH	TEB	TS

REV	DATE	REVISION	BY	CH	APP	APP
10	9-30-97	ISSUED AS-BUILT FOR PC/M 96-013.	RV	RH	BAP	TS
9	3-21-97	ISSUED AS-BUILT FOR PC/M 95-047.	RV	RH	BAP	TS
8	5-11-95	ISSUED AS-BUILT FOR PC/M 94-120	RV	RH	BAW	JCW
18	3/10/16	ISSUED AS-BUILT PER EC-DCR 285982.	RH	AFG	–	SB
17	5/1/13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-04-155).	RV	BB	–	PMB

QUALITY RELATED

POD

TURKEY POINT NUCLEAR UNITS 3 & 4

CONTROL SYSTEM DIAGRAM

ROD CONTROL SYSTEM

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-D-12A

SYS –

SHEET 1

REV 18

5610-T-D-12A_SH 1.DWG

Question # 57

Given the following:

- Unit 4 RCS is in MODE 6.
- Core offload is in progress.
- N-4-31, Source Range Instrument, is OOS.
- N-4-32, Source Range Instrument, is providing audible count rate.

Subsequently:

- The Refueling SRO communicates to the Control Room that he can NO longer hear the audible count rate in containment.

Which one of the following completes the statements below?

The crew __(1)__ required to suspend Core alterations.
Gammametrics Excore Nuclear Instruments __(2)__ capable of providing an audible count rate.

A. (1) is
(2) are

B. (1) is
(2) are NOT

C. (1) is NOT
(2) are

D. (1) is NOT
(2) are NOT

Answer Analysis

Discussion: Per Technical Specification 3.9.2 LCO, one Primary and one of the other three instruments that monitor the source range are required to move irradiated fuel. Audio count rate is required to be operable in both the containment and the control room. The Gamma Metrics Instrument are NOT capable of providing and AUDIBLE count rate, it can ONLY be provided from the SRNIs N-31 and N-32.

A. (1) Part 1 correct.
   (2) Incorrect, plausible since the Source Range Nis are the ONLY instruments that provided audible count rate, and Gammametrics Instruments in some cases can replace the Tech Spec function of Source Range Nuclear Instruments .

B. Correct, see discussion above

C. (1) Part 1 incorrect, plausible since both Gammametrics Channels remain OPERABLE.
   (2) Incorrect, plausible since the Source Range Nis are the ONLY instruments that provided audible count rate, and Gammametrics Instruments in some cases can replace the Tech Spec function of Source Range Nuclear Instruments .

D. (1) Part 1 incorrect, plausible since both Gammametrics Channels remain OPERABLE.
   (2) Part 2 correct.

Question Number: 57

Tier: 2 **Group:** 2

K/A: **015AK3.03 Knowledge of the effect that a loss or malfunction of the NIS will have on the Fuel Handling system:**

Importance Rating: **2.7**

10 CFR Part 55: 41.7

10 CFR 55.43.b **:** N/A

K/A Match: **RO question. Tech Spec aspect is above the line knowledge. Not HCL.** The K/A is matched as the applicant must evaluate plant conditions, identify the proper NIS and Containment indications and control functions and operator action required for the event.

SRO Justification: N/A

Technical References: 4-ONOP-059.5, Source Range NI Malfunction
Lesson Plan PTN 69000104

Proposed references to be provided: None

Learning Objective: PTN 6902104 OBJ. 13

Cognitive Level: Higher _

Lower X

Question Source: New X
Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

Immediate action per 4-ONOP-59.5

4.4 Modes 3, 4 and 5 - with reactor trip breakers in the open position/control rod drive system NOT capable of rod withdrawal.

4.4.1 Malfunction of ONE channel:

1. NONE

4.4.2 Malfunction of BOTH channels:

1. NONE

4.5 Mode 6 - Refueling

4.5.1 Malfunction of ONE channel:

1. Verify at least 2 out of 4 Source Range and Backup NIS (Gamma Metrics) channels are operable, with one Source Range having audible count rate in the Control Room and Containment.

   a. **IF** the above requirement is not met, **THEN** suspend all operations involving core alterations **OR** positive reactivity changes.

4.5.2 Malfunction of BOTH channels:

1. Suspend all operations involving Core Alterations.

2. Suspend all operations involving positive reactivity changes.

5.5 Mode 6 - Refueling

5.5.1 Malfunction of ONE channel:

1. Switch the AUDIO COUNT RATE CHANNEL SELECTOR to the operable source range.

2. **IF** applicable, **THEN** notify plant personnel of erroneous Containment Evacuation Alarm.

3. Place LEVEL TRIP switch on failed channel in BYPASS position.

4. Place HIGH FLUX AT SHUTDOWN switch on failed channel in BLOCK position.

5. Switch an NIS RECORDER to the operable source range.

6. **IF** one Source Range having audible count rate in the Control Room and Containment, **AND** 2 out of 4 NIS (NSSS Source Range and Gamma Metrics) Channels are not operable, **THEN** verify RCS boron concentration is greater than or equal to the required boron concentration at least once per 12 hours.

7. Monitor Backup NIS (Gamma Metric) Source Range Count Rate.

5.5.2 Malfunction of BOTH channels:

1. **IF** applicable, **THEN** notify plant personnel of erroneous Containment Evacuation Alarm.

2. Place LEVEL TRIP switches on failed channels in BYPASS position.

3. Place HIGH FLUX AT SHUTDOWN switches on failed channels in BLOCK position.

4. Switch NIS RECORDERS to the intermediate ranges.

5. Verify RCS boron concentration is greater than or equal to required concentration at least once per 12 hours.

6. Monitor Backup NIS (Gamma Metrics) Source Range Count Rate.

Question # 58

Given the following:

- Unit 3 is in MODE 5.
- Pressurizer Bubble is being established per 3-GOP-503, Cold Shutdown to Hot Standby.

Subsequently:

- The 3A 4kV buss DE-ENERGIZES due to an electrical fault.

Which one of the following identifies ALL of the PZR Heaters that will remain ENERGIZED?

A. ONLY Control Group Heater

B. ONLY Backup Heater Group A

C. ONLY Backup Heater Group B

D. Backup Heater Groups A and B.

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. Incorrect, Plausible if the candidate does not recall the power supply for the Control HTRS and assumes it is the D 480V LC which is powered from the B 4160V Bus

B. Incorrect, Plausible if the candidate thinks the B/U HTRS are A powered and the Control HTRS are B powered.

C. Correct, Backup Heater Bank 3B is provided power from the 3B 4160V Bus

D. Incorrect, Plausible if the candidate thinks the B/U HTRS are B powered and the Control HTRS are A powered.

Question Number: 58

Tier: 2 **Group:** 2

K/A: 011 (SF2 PZR LCS) Pressurizer Level Control
011K2.02 Knowledge of bus power supplies to the following: PZR heaters

Importance Rating: 3.1

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 5610-E-855

Proposed references to be provided: None

Learning Objective: PTN 6902109 OBJ. 4

Cognitive Level: Higher _
Lower _X

Question Source: New X_
Modified Bank _
Bank _

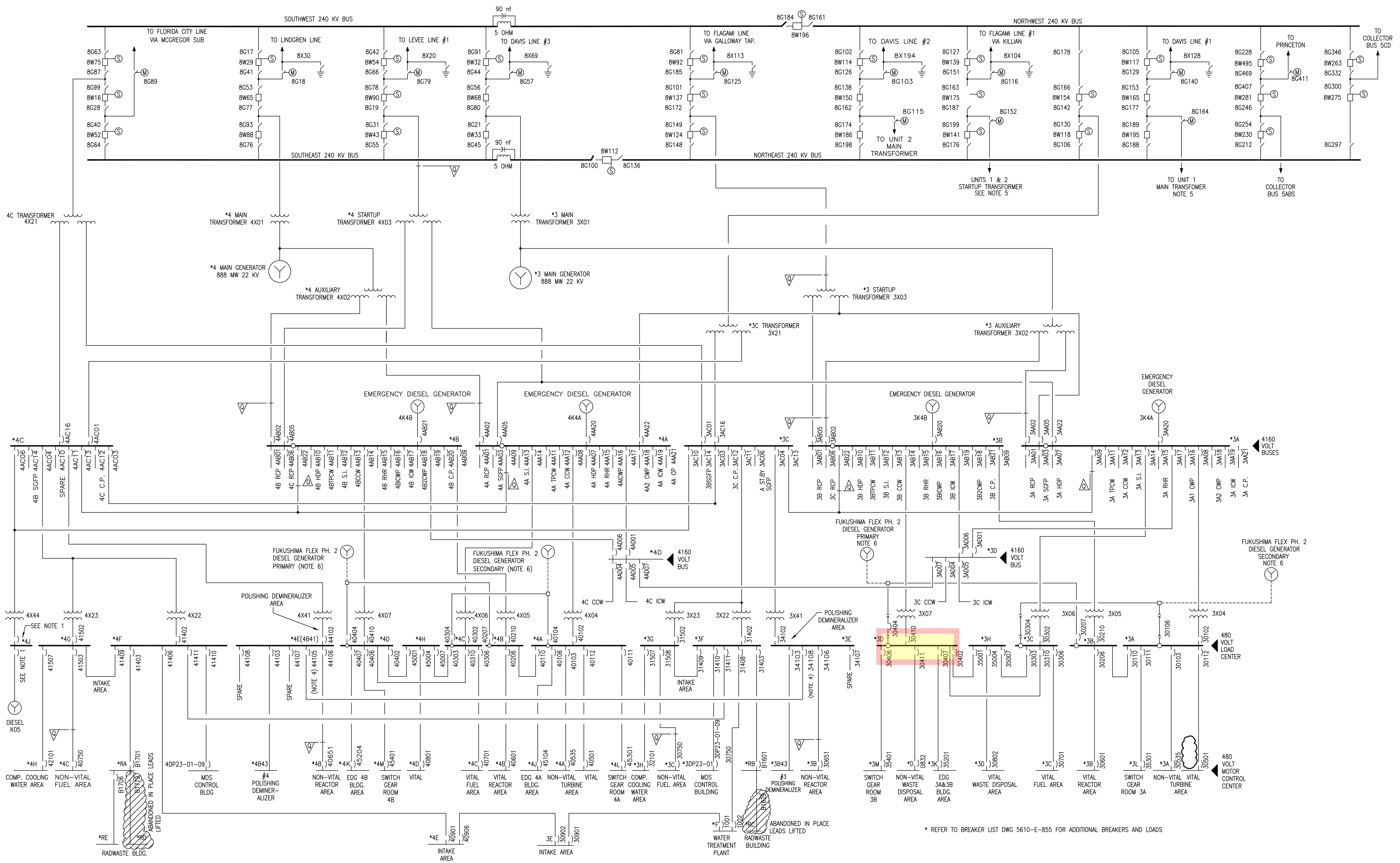
Question History: New

Comments: Screenshot of development references as applicable

INPUT DATA FILE PTP.BL.123   L.C. 3D (480V AC) (3B04)

CODE	HEATER BREAKER	SCHEMATIC LOCATION PNL SCHED.	EQUIP. NO. CKT DESC	EQUIPMENT AND OR LOCATION		BREAKER NUMBER
502	LP34A(3P04)-60 4160V SWGR ROOM	E-5	3S7	3B CONTROL ROD POWER SUPPLY MG SET 3B	(PC/M 04-117)	30401
253		E-5	3B50	LC-3H ELEC. EQUIP. ROOM AREA	(PC/M 87-258)	30402
	LP34A(3P04)-58 4160V SWGR ROOM	E-5	3P214B	3B CONTAINMENT SPRAY PUMP 3B		30403
709		5613-E-28 SH. 97D		PORTABLE DIESEL GENERATOR - FUKUSHIMA FLEX PHASE 2 - PRIMARY	(EC 279532)	30404
		E-5	H-1	TURBINE GANTRY CRANE		30405
249		E-5	3B54	FEED TO MCC 3M (SWITCHGEAR ROOM 3B) (PC/M 90-070)		30406
248		5613-E-5	3B52	MCC 3K EDG. 3 BLDG. AREA (PC/M 87-264)		30407
		E-5	3B13	PRESSURIZER BACK-UP HTR 3B		30408
		E-5		POTENTIAL TRANSF. COMPART & AUTO TRANSFER RELAY		30409
		E-5	3X07	MAIN FEED FROM L.C. TRANSFORMER 3D MAIN BREAKER		30410
127		E-5	B08	NON-VITAL FEED TO MCC NVD (WASTE DISPOSAL) (PC/M 86-095)		30411
192				SPACE (PC/M 88-530)		30412

CODES: NAB-NOT AS BUILT
RES-RESERVED



* REFER TO BREAKER LIST DWG 5610-E-855 FOR ADDITIONAL BREAKERS AND LOADS

NOTES:

1) SEE 5610-E-1522 FOR ADDITIONAL INFORMATION
2) DELETED
3) DELETED
4) 480V TIE-LINE BETWEEN LOAD CENTERS 3E & 4E IS A TEMPORARY MAINTENANCE TIE AND ITS USE IS CONTROLLED BY PROCEDURE 3/4-NOP-006.
5) NOTIFY NPS PRIOR TO COMMENCING WORK ON THIS SYSTEM.
6) PORTABLE DIESEL GENERATOR TO BE CONNECTED IN RESPONSE TO A BEYOND DESIGN BASIS EVENT PER FSG-004.

REV	DATE	REVISION	BY	CH	APP	APP
71	3/19/13	ISSUED AS-BUILT PER EC 249331 (ITOP 13-03-150).	RH	BB	–	ERJ
70	02-09-12	ISSUED AS-BUILT PER EC 242437 (ITOP 13-02-034)	RH	RV	–	PMB
69	12-08-12	ISSUED AS-BUILT PER EC-DCR 277655.	RV	BB	–	MA
68	08-30-12	ISSUED AS-BUILT PER EC 246916 (PC/M 06-167) .	RV	BB	DA	JD
79	12/1/16	ISSUED AS-BUILT PER EC 247019	DLM	RH	–	RJT
78	5/5/16	ISSUED AS-BUILT PER EC-DCR 286454.	LR	RH	–	SB

REV	DATE	REVISION	BY	CH	APP	APP
77	4/11/16	ISSUED AS-BUILT PER EC 279533..	RH	AFG	–	BB
76	2/9/16	ISSUED AS-BUILT PER EC-DCR 285774.	RH	AFG	–	SB
75	11/7/15	ISSUED AS-BUILT PER EC 279532	DLM	RH	–	BB
74	06-13-13	ISSUED AS-BUILT PER EC 246917 (ITOP 13-06-052).	RV	RH	–	PMB
73	04-02-13	ISSUED AS-BUILT PER EC-DCR 278017.	RV	BB	–	DBJ
72	3-29-13	ISSUED AS-BUILT PER EC 246904 & INCORP. CRN-017 (ITOP 13-03-168).	RH	RV	–	TK

SAFETY RELATED | POD

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

OPERATING DIAGRAM ELECTRICAL DISTRIBUTION

FLORIDA POWER & LIGHT

DRAWING NUMBER: 5610-T-E-1591

SYS: –

SHEET: 1

REV: 79

5610-T-E-1591_SH 1.DWG

Question # 59

Given the following:

- ANN H1/1, SFP LO ALARM, alarms.
- LEVEL LI-3-651, Spent Fuel Pit Level, indicates 55' 6 "

Which one of the following completes the statements below?

Normal Spent Fuel Pool cooling __(1)__ still be effective.

The MINIMUM number of inches of SFP makeup necessary to comply with Technical Specification 3.9.11 WATER LEVEL, STORAGE POOL is __(2)__inches.

A. (1) can
(2) 18

B. (1) can NOT
(2) 16

C. (1) can NOT
(2) 18

D. (1) can
(2) 16

Answer Analysis

Discussion: Spent Fuel Pool minimum level by TSs is 56 ‘ 10 inches. Normal flow is 2000 gpm. Level CAN be lowered to 55’ 2” with 2000 gpm before vortexing starts to become a concern. Question tests knowledge of when loss of level can cause vortexing <u>and impact SFP cooling</u>.

A. (1) Correct
(2) Incorrect, Plausible because a normal level of 57’ would be 18 inches

B. (1) Incorrect - SFP cooling can still be effective at specified level (55, 6”’). Plausible because it is less than required by TSs. 55’, 2” inches is the start of Vortex concerns. (See NOP – 033)
(2) Correct: SIXTEEN inches of makeup are needed for TS 3.9.11

C. (1) Incorrect, see B(1)
(2) Incorrect, see A(2)

D. Correct, see discussion above

Question Number: 59

Tier: 2 **Group:** 2

K/A: **033AA2.03 Ability to (a) predict the impact of Abnormal Spent Fuel Pool Water level or loss of water level on Spent Fuel Pool cooling; and (b) based on the prediction use procedures to correct, control, or mitigate the consequences:**

Importance Rating: **3.1 -10 CFR Part 55:** 41.5

10 CFR 55.43.b : N/A

K/A Match: **RO systems question (T2, G2)** The K/A is matched as the applicant must evaluate plant conditions, identify the impact of low water level on the ability to cool the SFP, Verify the low level will NOT lead to vortexing. and operator action (amount of water addition) required to mitigate the event / Restore level to compliance with the Tech Spec.

SRO Justification: N/A

Technical References:

H 1/1, SFP LO LEVEL
3-NOP-033 SFP System Operation
UFSAR section x
Lesson Plan PTN 6910141

Proposed references to be provided: None

Learning Objective PTN 6910141 OBJ. 7

Cognitive Level: Higher X

Lower _

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

5.8 Draining Spent Fuel Pool Level Below Tech Spec Minimum for Maintenance Activities

1. **ENSURE** applicable prerequisites have been satisfied.

2. **ENSURE** all fuel movement AND crane operations in U3 SFP are stopped during duration of the SFP draindown.

3. **ENSURE** U3 SFP ventilation is IN SERVICE.

4. **VERIFY** U3 Transfer Canal is drained AND available to receive water.

5. **VERIFY** there is an approved 10CFR 50.59 evaluation to lower U3 Spent Fuel Pool below the Tech Spec Minimum.

6. **DETERMINE** the desired level from the EC, and **CALCULATE** the volume of water to be sluiced. _______________ gallons.

7. **ENSURE** the 3B SFP Cooling Pump and the 3B SFP HX are IN SERVICE.

8. **ADJUST** 3-927, SFP HX 3E208B OUTLET ISOLATION VALVE, to maintain flow between 1600-1700 GPM as indicated on FI-3-1475A, SFP COOLING SUPPLY FLOW INDICATOR.

9. Sluice the required water volume from the Spent Fuel Pool to the Transfer Canal as follows:

CAUTION

3-927, SFP HX 3E208B OUTLET ISOLATION VALVE, will need to be adjusted to maintain SFP flow less than 1600 GPM to prevent vortexing if SFP level lowers to less than 54' 10".

NOTE

It is preferred to **NOT** raise level in the Transfer Canal above the bottom of the Keyway to prevent inadvertent equalization of the DP across the Keyway.

A. **OPEN** 3-12-029, SFP PMP SUCT VLV FROM FUEL TRANSFER CANAL.

B. **UNLOCK** and **OPEN** 3-12-028, DRN VLV FUEL TRANSFER CANAL - OUTSIDE CNTMT.

5.8 Draining Spent Fuel Pool Level Below Tech Spec Minimum for Maintenance Activities (continued)

11. (continued)

G. WHEN the SFP reaches the required level OR #1 WHT reaches 75%, THEN:

(1) **CLOSE** 3-918A, B SFP COOLING WTR. PP. & EMERGENCY PP. SUCTION X-TIE ISOLATION VL.

(2) **CLOSE** and **LOCK**, 3-12-30, DRN VLV FROM FUEL TRANSFER CANAL TO WHT.

(3) **CLOSE** 3-12-029, SFP PMP SUCT VLV FROM FUEL TRANSFER CANAL.

12. **RECORD** the SFP temperature and water level hourly during the interval when SFP water level is less than Technical Specification range (less than 56 feet 10 inches) using Data Sheet 1, Hourly SFP Water Level and Temperature Checks.

A. IF the level is **NOT** restored within 7 days, THEN **NOTIFY** the NRC within the next 24 hours. (TS 3/4.9.11).

13. WHEN maintenance is complete AND it is desired to transfer the water from the U3 Transfer Canal to the U3 SFP, THEN **PERFORM** the following:

A. **NOTIFY** RP Shift Supervisor of intent to drain Transfer Canal and fill the SFP.

B. **VERIFY** the Emergency SFP Pump is available.

C. **VERIFY** the 3B SFP Cooling Pump is aligned to the 3B SFP HX with the B loop in operation.

Question # 60

Given the following:

- Unit 4 is in MODE 6.

Which one of the following completes the statements below?

IAW 4-NOP-033, Spent Fuel Pit Cooling System, prior to core off-load a MINIMUM of __(1)__ SFP Heat Exchanger(s) is(are) required to be in service.

IAW 4-OP-038.1, Preparation for Refueling Activities, fuel off-load to the Spent Fuel Pool (SFP) shall be suspended if SFP temperature reaches __(2)__ as a MINIMUM.

A. (1) 1
(2) 126°F

B. (1) 1
(2) 140°F

C. (1) 2
(2) 126°F

D. (1) 2
(2) 140°F

Answer Analysis

A. (1) Part 1 incorrect, plausible since this is the normal SFP cooling lineup.
(2) Part 2 incorrect, plausible since this is the SFP temperature that triggers the requirement to use fall protection IAW 0-ADM-033

B. (1) Part 1 incorrect, plausible since this is the normal SFP cooling lineup.
(2) Part 2 correct.

C. (1) Part 1 correct.
(2) Part 2 incorrect, plausible since this is the SFP temperature that triggers the requirement to use fall protection IAW 0-ADM-033

D. Correct. IAW 4-NOP-033, Both SFP Pumps and Heat Exchangers must be in service prior to core offload. 4-OP-38.1 requires to stop the core offload if SFP temperature rises to 140°F.

Question Number: 60

Tier: 2 **Group:** 2

K/A: 034 (SF8 FHS) Fuel Handling Equipment
034G2.1.32; Ability to explain and apply system limits and precautions.

Importance Rating: 3.8

10 CFR Part 55: 41.10

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 4-NOP-033, 4-OP-038.1

Proposed references to be provided: None

Learning Objective: PTN 6910141 OBJ. 7

Cognitive Level: Higher _

Lower _X

Question Source: New X
Modified Bank _
Bank _

Question History: New

Comments: Screenshot of development references as applicable

1.0 PURPOSE

This procedure provides instructions for startup, normal operation, shutdown, and infrequent operations of the Spent Fuel Pit Cooling System, including the Spent Fuel Pit Cooling Loop, the Spent Fuel Pit Purification Loop, the Spent Fuel Pit Skimmer Loop, and the RWST Purification Loop.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

1. Parallel operation (two pumps and two heat exchangers) can only be performed if the SFP impingement plate is installed.

2. When working around the Spent Fuel Pit:

   - All loose items (pens, pencils, etc.) shall be removed from the person.
   - Eyeglasses, TLDs, dosimeters, etc. shall be securely attached to the person.
   - All tools shall be attached to a line and secured.
   - Upon entering the Spent Fuel Pit area, a personal flotation device or a safety belt shall be worn, whenever the possibility of falling into the water exists.

3. Spent Fuel Pit area radiation monitors should be in service anytime irradiated fuel is in the storage pit.

4. Spent Fuel Pit area ventilation should be in service anytime irradiated fuel is in the storage pit.

5. Prior to Core Offload, both SFP heat exchangers and pumps must be placed in parallel operation. During and after full core offload, the total cooling capacity of both heat exchangers are required to meet the total decay heat load in the SFP, and ensure water temperature in the SFP does **NOT** exceed the administrative limit of 140°F. This precaution applies to Core Offloading after initial Cycle 27 core operation at EPU conditions.

4.7 If Control Rod/Upper Internals movement activities are stopped, complete Attachment 3 prior recommencing Control Rod/Upper Internals movement. [Commitment - Step 2.3.3]

4.8 Prior to the start of fuel movement in the reactor vessel, the reactor shall be subcritical for the conditions prescribed in the cycle-specific reload PC/M. (This only applies to fuel movement, not control rod movement.)

4.9 Prior to start of fuel movement from the reactor vessel, the emergency SFP pump shall be connected to a reliable power source and ready for operation.

4.10 The Spent Fuel Pool temperature shall be less than 100°F at the time of offload. [Commitment - Step 2.3.4]

4.11 Fuel offload to the SFP SHALL be suspended if Spent Fuel Pool temperature is greater than or equal to the administrative limit of 140°F. Fuel offload shall not resume until the cause is understood, evaluated, and corrected via the Condition Report process. [Commitment – Steps 2.3.6 and 2.3.7]

4.12 The fuel assembly transfer rate from the reactor to the SFP shall be limited to a cumulative average of six fuel assemblies per hour commencing at the start of the offload. In addition, no more than eight fuel assemblies shall be transferred during a single hour. More restrictive limits may be imposed by Engineering based on specific fuel cycle and plant conditions. [Commitment – Step 2.3.6.]

4.13 Shift Manager approval SHALL be obtained for all sequence changes and any changes in Shuffle Data Sheets, (0-ADM-556, Fuel Assembly and Insert Shuffles, Attachment 6).

4.14 Communications:

4.14.1 Verbatim repeat-backs and phonetic alphabet SHALL be used during all communications.

4.14.2 The following, as a minimum, SHALL be communicated between stations for each fuel assembly:

1. Step number from the Shuffle Data Sheet.
2. Specific location in the reactor or SFP as applicable.

Question # 61

Which one of the following completes the statements below?

ALSO PROVIDED AS A REFERENCE IN LARGER FORMAT

On the main feedwater flow control valve control stations PV1 indicates __(1)__, and MV indicates __(2)__.

A. (1) measured SG NR level
(2) actual valve position

B. (1) measured SG NR level
(2) controller demand

C. (1) desired SG level
(2) actual valve position

D. (1) desired SG level
(2) controller demand

Answer Analysis

A. (1) Correct
   (2) Incorrect, Credible as one would expect actual valve position to be at or near the measured and/or desired position and level if normal level was about 50%

B. Correct

C. (1) Incorrect Credible because the letters PV could be interpreted many ways.
   (2) Incorrect, see A(2).

D. (1) Incorrect See C(1)
   (2) Incorrect, see A(2).

Question Number: 61

Tier: 2 **Group:** 2

K/A: **035AA3.01 Ability to monitor automatic operation of the S/G including:** Steam Generator Water Level control

Importance Rating: **4.0 10 CFR Part 55:** 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must identify the proper control system features of FCV-488 to determine proper system response.

SRO Justification: N/A

Technical References: Lesson plan PTN 6902111A

Proposed references to be provided: None

Learning Objective: PTN 6902111A OBJ. 1

Cognitive Level: Higher _

	Lower	X
Question Source:	New	X
	Modified Bank	_
	Bank	_
Question History:	New	
Comments:	Screenshot of development references as applicable	

Yokogawa Controllers

Various Info: Instrument number, Loop1 is display page designator (1 of 5 main pages)
PF =Programmable Function

Fail Alarm –Red indicates controller problem.

ALM or Alarm –Yellow indicates abnormal condition of and input or output.

PV1 = Process Variable or actual S/G Level
SV1 = Setpoint Variable or desired S/G Level

This 'Tag' will always say 'CLOSE' in PTN Application of this controller.

The "PF" key is used to swap from primary to back up controller and back by depressing and holding for 6 sec. "PF" key will back lit when in active control.

There are 11 pages of information which can be accessed by depressing blue page key. Ops only has access to page shown.

LP 6902111A

MV1 = Manipulated Variable, output signal to valve positioner.

Question # 62

Given the following:

- A release of a Waste Monitor Tank is in progress.
- ANN H1/4, PRMS HI RADIATION, alarms.
- R-18, Waste Disposal Liquid Effluent PRMS Channel, HIGH alarm is lit.

Which one of the following completes the statements below?

IAW 3-ONOP-067, Radioactive Effluent Release, the crew is required to ______.

A. LOCALLY trip the in-service Waste Monitor Tank discharge pump.

B. LOCALLY close RCV-018, Liquid Waste Effluent Isolation Valve.

C. Verify that the in-service Waster Monitor Tank discharge pump AUTOMATICALLY trips.

D. Verify that RCV-018, Liquid Waste Effluent Isolation Valve AUTOMATICALLY closes.

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. Incorrect. Plausible because this would stop flow but incorrect because high radiation will isolate the flowpath with RCV-018

B. Incorrect. Plausible because this isolates the flowpath but manual isolation is not required. The high radiation signal will isolate RCV-018

C. Incorrect. Plausible because this would stop discharge flow and because if the candidate knows there is an automatic action but is unsure what that action is they could easily choose this response.

D. Correct. RCV-018 automatically closes on a high alarm.

Question Number: 62

Tier: 2 **Group:** 2

K/A: 068 (SF9 LRS) Liquid Radwaste

068AK6.10 ; Knowledge of the effect of a loss or malfunction on the following will have on the Liquid Radwaste System:

Radiation monitors

Importance Rating: 2.5

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-ONOP-067

Proposed references to be provided: None

Learning Objective: PTN 6902242 OBJ. 6

Cognitive Level: Higher _

Lower _

Question Source: New _

Modified Bank _

Bank X

Question History: This question was used on the 2016 PTN NRC exam.

.

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If more than one high radiation event is occurring, the operator should prioritize actions to minimize offsite dose.

NOTES

- *Prioritization should include consideration of release rate, size of leak, isolable or NOT, etc.*
- *Step 3 RNO actions should be performed in the determined order of priority.*

3 **Check PRMS High Alarm - OFF**

- Check R-11 **AND** R-12 High Alarms - OFF
- Check R-17A **AND** R-17B High Alarms - OFF
- Check R-14 High Alarm - OFF
- Check R-18 High Alarm - OFF
- Check R-20 High Alarm - OFF

RNO: Perform the following:

* **IF** R-11 **OR** R-12 High Alarm is ON, **THEN** go to Step 16.

* **IF** R-17A **OR** R-17B High Alarm is ON, **THEN** go to Step 29.

* **IF** R-14 High Alarm is ON, **THEN** go to Step 42.

* **IF** R-20 High Alarm is ON, **THEN** perform 3-ONOP-041.4, EXCESSIVE REACTOR COOLANT SYSTEM ACTIVITY, while continuing with this procedure.

* **IF** R-18 High Alarm is ON, **THEN** perform the following:

  a. Verify RCV-018 - Closed.

  b. **IF** a Liquid Release is in progress, **THEN** terminate the release.

  c. Inform the Shift Manager of R-18 alarm.

  d. Determine and correct the cause of the R-18 high alarm before commencing another liquid release.

Question # 63

If a core exit thermocouple (CET) location reaches 2350°F, the associated QSPDS display will indicate __(1)__. The background color of the display will be __(2)__.

A. (1) 2350°F
(2) green

B. (1) 2300°F
(2) red

C. (1) 2350°F
(2) red

D. (1) 2300°F
(2) green

Answer Analysis

Discussion: The maximum range of a CET is 2300. If a value on QSPDS alarms, the background turns red. The alarm occurs at 700°F.

A. (1) Incorrect. It is credible that a display will accurately display a process variable.
(2) Incorrect. It is credible because this is the normal condition of the display.

B. Correct

C. (1) Incorrect, see A(1)
(2) Correct

D. (1) Correct
(2) Incorrect. See A(2)

Question Number: 63

Tier: 2 **Group:** 2

K/A: **RO Question. (T2 G2) 017K4.03: Knowledge of ITM system design feature(s) and/or interlocks(s) which provide for:** Range of temperature indication.

Importance Rating: **3.1 10 CFR Part 55:** 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must pull knowledge of expected accident temperature range required by CETs to monitor and respond to core cooling events.

SRO Justification: N/A

Technical References: 5614-M-313, PTN 6918171

Proposed references to be provided: None

Learning Objective: PTN 6918171 OBJ. 6

Cognitive Level: Higher _

Lower X

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

TAG NUMBER ASSOCIATE TAG VENDOR TAG CONTROL CLASS SYSTEM	DESCRIPTION	PROCESS RANGE ACTUATE RANGE INSTRUMENT ACC	PRO SP & TOL PRO RESET & TOL ACT SP & TOL ACT RESET & TOL FUNCTION ACTION	ORIG. DOCUMENT	REF. DOC.	SHT. NO.	REMARKS	REV
4PDSA NOTE 4	CONTINUATION		16 PSIG * LO INDICATION TRIPS ON DECR				* OUT-OF-RANGE SP. TYP. RCS PRESSURE VALUE SEE NOTES 1a & 4 PC/M 03-109	52
4PDSA NOTE 5 4 041	ICCS REACTOR VESSEL LEVEL OUTLET PLENUM	0-100% SPAN	LESS THAN 100% SPAN* LO ALARM TRIPS ON DECR		16081-ICE-3219 4-ARP-097.CR	 AN-A-2/2	* ABOVE THE FUEL ALIGNMENT PLATE (FAP). SEE NOTES 1e, 1g & 5 PC/M 03-109	52
4PDSA NOTE 5 4 041	ICCS REACTOR VESSEL LEVEL HEAD	0-100% SPAN	LESS THAN 100% SPAN LO ALARM TRIPS ON DECR		16081-ICE-3219 4-ARP-097.CR	 AN-A-2/2	SEE NOTES 1e, 1g & 5 PC/M 03-109	52
4PDSA NOTE 6 4 041	ICCS CORE EXIT THERMOCOUPLE A-8 TEMPERATURE	32-2300 DEG F 0-50.99 MVDC	2,295 DEG F * HI INDICATION TRIPS ON INC		SPEC-IC-022		* OUT-OF-RANGE SP. TYP. VALUE, SEE NOTES 1a & 6. PC/M 03-109	52
4PDSA NOTE 6	CONTINUATION		540 DEG. F LOW ALARM TRIPS ON DECR		16081-ICE-3219		NOTE 1d PC/M 03-109	52
4PDSA NOTE 6	CONTINUATION		VARIABLE HI ALARM TRIPS ON INC	PC/M 90-293	16081-ICE-3219		TYP. VALUE SEE NOTE 1f PC/M 03-109	52
4PDSA NOTE 6	CONTINUATION		1200 DEG F HI-HI ALARM TRIPS ON INC	PC/M 03-109	SPEC-IC-022		NOTE 1d PC/M 03-109	52

The color of the background will change as the quality of the input degrades.

Blue background the input or calculation is suspected of being out of range.

White backround the input or calculation has failed.

Black background with magenta lettering means the input is offscan.

Anything displayed in blue is suspect and should not be used without reviewing other supporting parameters.

Question # 64

Which one of the following completes the statements below?

The obstruction of the 3A2 Traveling Screen affects the operation of the __(1)__ ICW Pump. The effect of this screen obstruction on ICW is mitigated in __(2)__.

A. (1) 3A
(2) 3-ONOP-011, Screen Wash System/Intake Malfunction

B. (1) 3A
(2) 3-ONOP-019, Intake Cooling Water Malfunction

C. (1) 3B
(2) 3-ONOP-011, Screen Wash System/Intake Malfunction

D. (1) 3B
(2) 3-ONOP-019, Intake Cooling Water Malfunction

Answer Analysis

A. (1) Incorrect. Credible because there is nothing in the Circulating water pump number that makes it obvious which ICW pump is affected.
(2) Correct

B. (1) Incorrect. See A(1)
(2) Incorrect. Credible because other events that challenge ICW operation are in this procedure.

C. (1) Correct
(2) Correct

D. (1) Correct
(2) Incorrect. See B(2)

Question Number: 64

Tier: 2 **Group:** 2

K/A: 075 (SF8 CW) Circulating Water

075AK1.01; Knowledge of the physical connections and/or cause effect relationships between the circulating water system and the following systems: SWS

Importance Rating: 2.5

10 CFR Part 55: 41.2 to 41.9

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant must know how circ water and ICW share intake screens.

SRO Justification: N/A

Technical References: PTN 6902152 OBJ. 3

Proposed references to be provided: None

Learning Objective:

Cognitive Level: Higher _

Lower X

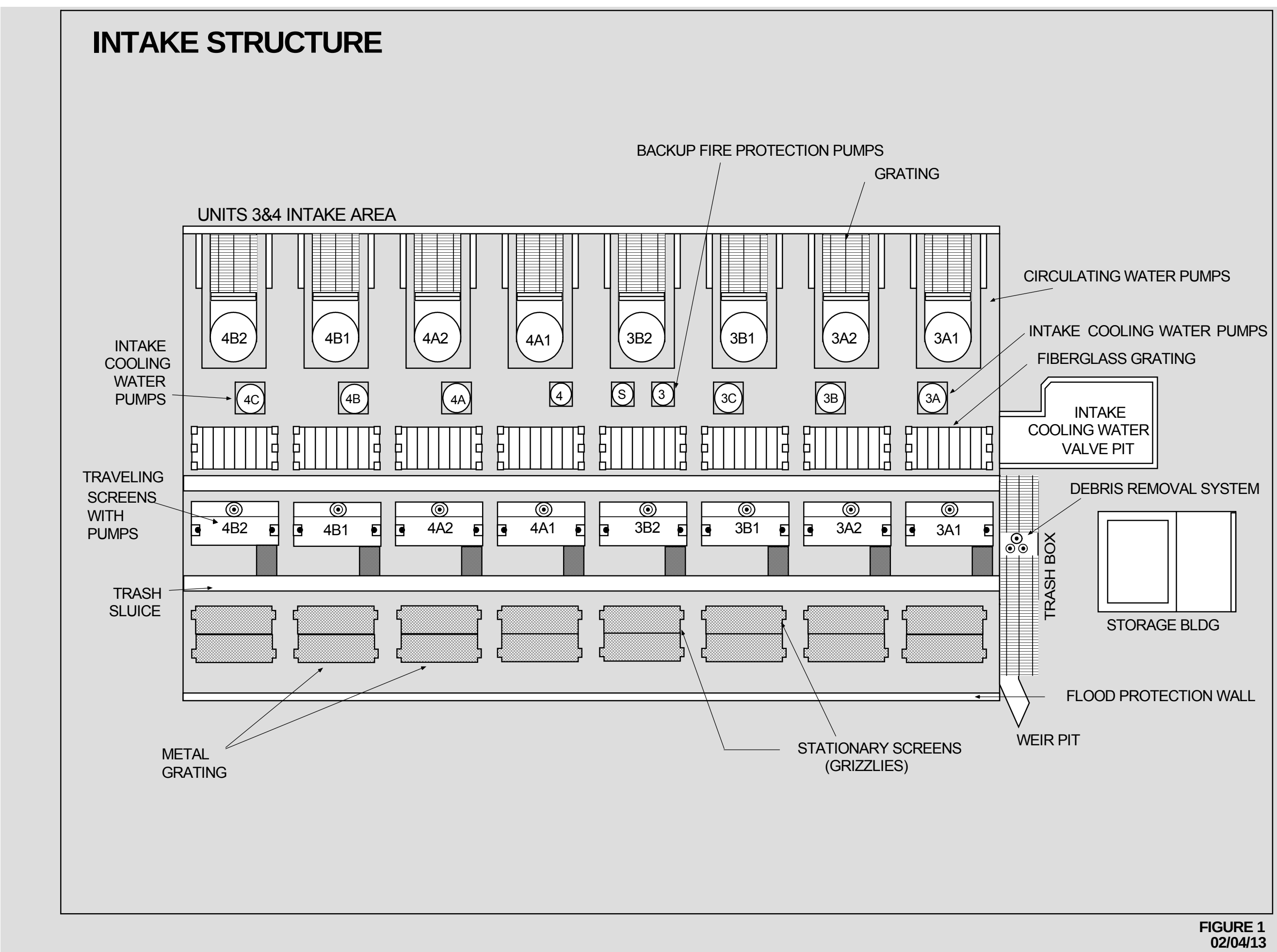
Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable



STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***If the cause of the Intake Cooling Water malfunction is determined to be due to high differential pressure on the traveling screens, then 3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION, should be used.***
- ***If an Intake Cooling Water Pump is stopped in this procedure, and the reason for stopping the pump has NOT been corrected, that pump is NOT available for starting in subsequent procedure steps.***
- ***Monitoring Main Generator RTDs is required if TPCW flow or temperature is changed due to the effect on Main Generator hydrogen leakage. An increase in hydrogen leakage is expected if the gas temperature to rotor temperature gradient increases. (Reference CR 2008-803)***

NOTE

Foldout Page shall be monitored throughout this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Verify All Intake Cooling Water Pump Alarms - OFF** • I 4/1, ICWP A/B/C MOTOR OVERLOAD • I 4/2, ICWP A/B/C TRIP • I 4/3, ICWP A/B/C MOTOR BRG HI TEMP	Perform the following: 1. Have operator check pump(s) locally. 2. Determine affected intake cooling water pump. 3. Start standby intake cooling water pump. 4. Stop affected intake cooling water pump.
2	**Check Traveling Screens - CLEAN** • Alarm I 3/3, TRAVELING SCREEN HI ΔP - OFF • Traveling Screen DP - LESS THAN 25 INCHES OF WATER	Go to 3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION.

Question # 65

Which one of the following completes the statements below?

IAW 0-ADM-033, PTN Industrial Safety Program, the hazard to personnel due to the type of fire suppression installed in the Cable Spreading Room would be ______.

A. electrical shock

B. respiratory problems

C. flood hazard

D. hearing damage

Answer Analysis

Discussion:

A. Incorrect, plausible since other fire protection systems like sprinklers could constitute a risk of electrical shock where energized equipment is present.

B. Correct, Halon will cause irritation of the respiratory airways and oxygen displacement.

C. Incorrect, plausible since other fire protection systems like sprinklers could constitute a risk of flooding if proper drainage is not present.

D. Incorrect, plausible since a loud noise is generated during Halon discharge.

Question Number: 65

Tier: 2 **Group:** 2

K/A: **086AK5.04 Knowledge of Knowledge of the operational implications of:** Hazards to personnel as a result of fire type and methods of protection **as they apply to the Fire Protection System**

Importance Rating: **2.9 - 10 CFR Part 55:** 41.5

10 CFR 55.43.b :N/A

K/A Match: The K/A is matched as the applicant must evaluate suppression required for fires in the CSRs and identify the proper personnel hazard.

SRO Justification: N/A

Technical References: PFP-CB-30, MSDS, 0-ADM-033

Proposed references to be provided: None

Learning Objective: PTN 6902143 OBJ. 1 & 3

Cognitive Level: Higher

Lower x_

Question Source: New X

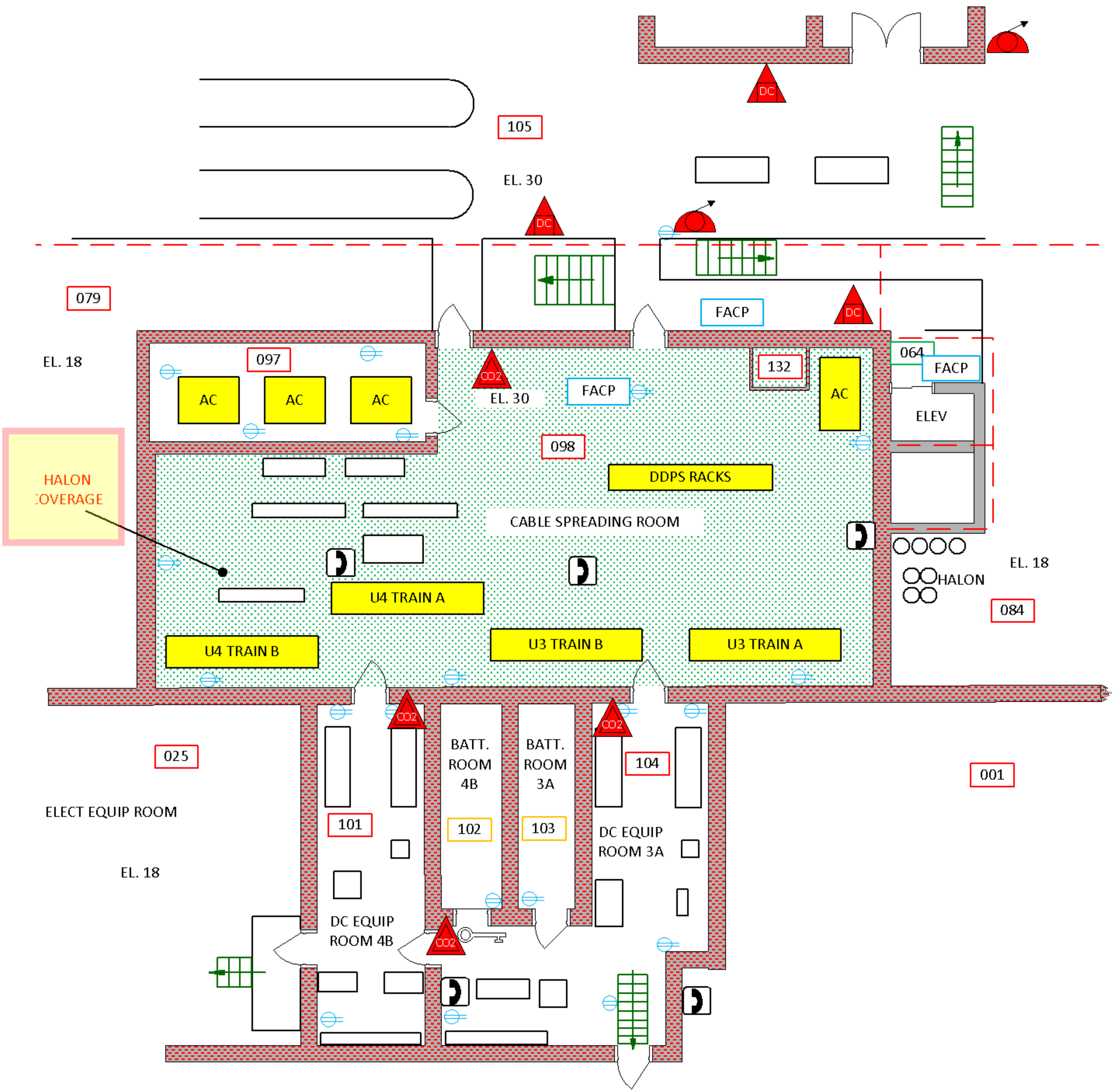
Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

PFP-CB-30
Unit 3&4 Control Building
Revision 1



Fire Barrier
Halon System Coverage
Sprinkler Coverage
Pre-Action Sprinkler Coverage
Deluge System Coverage
CO2 System Coverage
Dry Chemical Extinguisher
CO2 Extinguisher
Water Extinguisher
Fire / Equipment Locker
Hose Station
FD Floor Drain
Fire Hydrant
Fire Department Connection
Deluge Valve
FACP Fire Alarm Control Panel
Combustible Liquid
Deluge Actuation Station
Halon Actuation Station
Electrical Outlet
020 (LOW) 020 (MEDIUM) 020 (HIGH) CCDP
Electrical Hazard
Radiation
RCA Boundary
Phone / Page
Sliding Door
Locked Door
Card Access Door

Material Safety Data Sheet

Material Name: Recycled Halon 1301 **ID: KA003**

* * * Section 1 - Chemical Product and Company Identification * * *

Chemical Name: Bromotrifluoromethane in a pressurized container
Product Use: Extinguishing Fires
Synonyms: CC0111
Manufacturer Information

Kidde Aerospace
4200 Airport Drive, NW
Wilson, NC 27896

Phone: 252-237-7004

Emergency # 1-800-451-8346; 760-602-8700 (3E Company)

* * * Section 2 - Composition / Information on Ingredients * * *

CAS #	Component	Percent
75-63-8	bromotrifluoromethane	>99

Component Related Regulatory Information

This product may be regulated, have exposure limits or other information identified as the following: Bromofluorocarbons, Bromine compounds.

Component Information/Information on Non-Hazardous Components

This product is considered to be hazardous under 29 CFR 1910.1200 (Hazard Communication). This is a controlled product under the criteria specified in the Canadian Workplace Hazardous Materials Information System (WHMIS).

* * * Section 3 - Hazards Identification * * *

Emergency Overview

Warning. Asphyxiant. Inhalation of vapors of this product may affect the cardiovascular and central nervous system and may cause death. Skin or eye contact with the liquid will cause frostbite. Pressurized container may explode when exposed to heat or flame.

Potential Health Effects: Eyes

Contact with the liquid of this product will cause frostbite to the eyes.

Potential Health Effects: Skin

Contact with the liquid of this product will cause frostbite to the skin.

Potential Health Effects: Ingestion

Not a likely route of entry.

Potential Health Effects: Inhalation

Asphyxiant. The vapors of this product reduce oxygen available for breathing and are heavier than air. Inhalation of the vapors of the product causes central nervous system depression and affects the cardiovascular system. Symptoms include nausea, vomiting, irregular heartbeat, symptoms of drunkenness, disorientation, bluish skin color, suffocation, convulsions and possibly death.

HMIS Ratings: Health: 1* **Fire:** 0 **Physical Hazard:** 0
Hazard Scale: 0 = Minimal 1 = Slight 2 = Moderate 3 = Serious 4 = Severe * = Chronic hazard

* * * Section 4 - First Aid Measures * * *

First Aid: Eyes

Immediately flush eyes with water for at least 15 minutes, while holding eyelids open. Seek medical attention at once.

First Aid: Skin

Get medical attention. If frostbite or freezing occurs, immediately flush with plenty of lukewarm water (105-115°F; 41-46°C) for at least 15 minutes. Do not use hot water.

First Aid: Ingestion

If large amount is swallowed, get medical attention.

First Aid: Inhalation

Get medical attention. Remove the affected person immediately to fresh air.

Question # 66

Which one of the following completes the statements below?

IAW 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage,
a crew brief is required when transitioning from 3-EOP-E-0, Reactor Trip or Safety Injection, to ______.

A. 3-EOP-ECA-0.0, Loss of All AC

B. 3-EOP-FR-S.1, Response to Nuclear Power Generation/ATWS

C. 3-EOP-E-3, Steam Generator Tube Rupture

D. 3-EOP-E-1, Loss of Reactor or Secondary Coolant

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. **Incorrect.** This is incorrect because the transition brief is prohibited. According to 0-ADM-211 crew briefings should NOT be performed during critical evolutions that may delay event mitigation and cause further plant degradation such as transitioning from E- 0 to ECA-0.0. The first priority of the crew should involve the restoration of power to a Safeguards Bus. This is plausible because ECA-0.0 is a procedure to which a transition can be made from E-0, and the operator may incorrectly believe that a crew brief is required.

B. **Incorrect.** This is incorrect because the transition brief is prohibited. According to 0-ADM-211 crew briefings should NOT be performed during critical evolutions that may delay event mitigation and cause further plant degradation such as transitioning from E- 0 to FR-S.1 until the plant is stabilized. The first priority of the crew should be to stabilize the plant after the trip. This is plausible because FR-S.1 is a procedure to which a transition can be made from E-0, and the operator may incorrectly believe that a crew brief is required.

C. **Incorrect.** This is incorrect because the transition brief is prohibited. According to 0-ADM-211 crew briefings should NOT be performed during critical evolutions that may delay event mitigation and cause further plant degradation such as transitioning from E-0 to E-3. The first priority of the crew should involve actions to prevent ruptured SG release to the environment. This is plausible because E-3 is a procedure to which a transition can be made from E-0, and the operator may incorrectly believe that a crew brief is required.

D. **Correct.** According to 0-ADM-211, during the mitigation of an off normal or emergency event, it is imperative to conduct a crew briefing to ensure that the Control Room Team is aware of the plant status and event mitigating strategy. Briefings should be performed when operator actions to mitigate plant transients are NOT in progress. Briefings should NOT be performed prior to the initial review of CSFSTs is completed. According to 3-EOP-E-0, the monitoring of the CSFSTs is done prior to the Transition to E-1, and there will be no actions in progress on the transition. A Transition brief is required because the actions necessary to mitigate and stabilize the plant were taken in 3-EOP-E-0, Reactor Trip or Safety Injection.

Question Number: 66

Tier: 3 **Group:**

K/A: Conduct of Operations; 2.1.38, Knowledge of the station's requirements for verbal communications when implementing procedures.

Importance Rating: 3.7

10 CFR Part 55: 41.10

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 0-ADM-211

Proposed references to be provided: None

Learning Objective: PTN 6902320 OBJ. 2

Cognitive Level: Higher _

Lower _X

Question Source: New _

Modified Bank _

Bank X

Question History: Question was a new question and used on the 2013 PTN NRC Exam question #67.

Comments: Screenshot of development references as applicable

4.4 Briefings

1. During the mitigation of an off normal or emergency event, it is imperative to conduct a crew briefing to ensure the Control Room Team is aware of the plant status and event mitigating strategy.

A. The annunciator TIMED SILENCE switch may be used during crew briefings in the EOPs when the number of annunciators in alarm status impedes crew communications or recovery actions.

2. Briefings should be short and concise in duration, normally **NO** more than one to three minutes in length.

3. Briefings should be performed when operator actions to mitigate plant transients are **NOT** in progress.

A. Briefings should **NOT** be performed prior to the initial review of CSFSTs is completed.

B. If giving a brief hinders the event mitigation process, then a brief should **NOT** be performed during that time.

EXAMPLE

Transition from E-0 to ES-0.1 following a reactor trip without SI.

1. Procedure director announces the procedure transition to the Control Room team by performing an Update.
2. Crew performs specific actions of ES-0.1 as necessary to stabilize and control the plant following a reactor trip.
   - Ensure the Primary System Stabilizes at No-load conditions
   - Ensure the Secondary System Stabilizes at No-load conditions
3. STA completes initial review of the CSFSTs.

4. Crew briefings should **NOT** be performed during critical evolutions that may delay event mitigation and cause further plant degradation such as:

- Transitioning from E-0 to ES-0.1 until the plant is stabilized
- Transitioning to E-3, ECA-1.1, ECA-0.0, ECA-0.1, ECA-0.2, ECA-3 Series or ES-1.3
- Transitioning to any Red or Orange FRP

Question # 67

Given the following:

- Unit 3 is at 100%.

Subsequently:

- ANN A4/6, VCT HI/LO LEVEL, alarms.
- LI-3-115, VCT LEVEL, on VPA is at 82% and slowly rising.
- LI-3-112, VCT Level, on DCS is at 82% and slowly rising.
- PI-3-117, VCT pressure, on VPA is slowly lowering.

Which one of the following completes the statements below?

All of these are symptoms of ______.

A. a VCT Level instrumentation common reference leg failure

B. a spurious automatic VCT makeup

C. LCV-3-115A, VCT divert valve, failed in the VCT position

D. A failure of the DCS System

Answer Analysis

Discussion: These would be symptoms of common reference leg failure on the VCT. If pressure is dropping, actual level must be dropping, not rising, so both level channels must be wrong. The failure of channel 115 has caused the divert valve to divert all letdown away from the VCT.

A. Correct
B. Incorrect - An actual rise in level would make VCT pressure rise. Credible if the applicant thinks that two instruments that agree must be weighted more heavily than one channel.
C. Incorrect - An actual rise in level would make VCT pressure rise. See B
D. Incorrect - DCS has no control functions. See B

Question Number: 67

Tier: 3 **Category 1: Conduct of Operations**

K/A: **G2.1.45 Ability to identify and interpret diverse indications to validate the response of another indication**

Importance Rating: **4.3** **10 CFR Part 55:** 41.7

10 CFR 55.43.b **:**N/A

K/A Match: The K/A is matched as the applicant must evaluate a diverse parameter (VCT pressure) to validate level channel responses.

SRO Justification: N/A

Technical References: 3-ARP-097.CR.A, A4/6

Proposed references to be provided: None

Learning Objective: PTN 6902235 OBJ. 4

Cognitive Level: Higher X_ Needs to evaluate diverse indicaitons

Lower _

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

CAUSES:
1. Failed VCT level instrumentation
2. Loss of charging/letdown
3. RCS leak
4. VCT auto makeup malfunction

A4/6

VCT
HI/LO LEVEL

DEVICE:	SETPOINT:	LOCATION:
LT-3-115	HI - 81% LO - 12%	N/A

ALARM CONFIRMATION

1. **CHECK** LI-3-115, VCT LEVEL on VPA.
2. **CHECK** VCT level on LT-3-112 by at least one of the following methods:
   - **CHECK** LT-112 on DCS Chemical & Volume Control System display.
   - **DISPATCH** operator to check local reading on LI-3-112 in the Charging Pump Room.
   - **ADJUST** LC-3-112, VCT LEVEL CONTROLLER MANUAL/AUTO STATION, setpoint potentiometer until demand begins to indicate greater than zero, read the pot setting, then return to previous setting.

OPERATOR ACTIONS

NOTE

LT-3-112 and LT-3-115 share a common dry reference leg and a common wet variable leg. Failure of the common dry reference leg will cause a false high level.

1. IF LT-3-115 has failed high, THEN **PLACE** LCV-3-115A, L/D DIVERT FROM VCT TO HOLD-UP TANK, control switch to VCT position.
2. IF LT-3-112, VCT LEVEL TRANSMITTER, has failed HI, THEN **PERFORM** the following:
   A. **PLACE** LC-3-112, VCT LEVEL CONTROLLER AUTO-MANUAL STATION, in MANUAL.
   B. **MAINTAIN** VCT level between 17% to 37%.
3. IF LT-3-115 has failed low AND LI-3-112 is stable or increasing, THEN **TURN** RCS Make Up Control Switch to STOP.

Question # 68

Which one of the following completes the statement below?

IAW OP-AA-101-1000, Clearance and Tagging, a Configuration Control Clearance prohibits the use of ______tags.

A. Caution

B. Danger

C. Information

D. No-Tag, virtual

Answer Analysis

A. Incorrect, plausible since this is a type of ECO tag and the “Danger” type ECO tag is not allowed to be used for status control clearances.

B. Correct IAW 0-OP-AA-101-1000.

C. Incorrect, plausible since this is a type of ECO tag and the “Danger” type ECO tag is not allowed to be used for status control clearances.

D. Incorrect, plausible since this is a type of ECO tag and the “Danger” type ECO tag is not allowed to be used for status control clearances.

Question Number: 68

Tier: 3 **Group:**

K/A: Equipment Control; 2.2.14, Knowledge of the process for controlling equipment configuration or status.

Importance Rating: 3.9

10 CFR Part 55: 41.10

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: OP-AA-101-1000

Proposed references to be provided: None

Learning Objective: PTN 6902008 OBJ. 7

Cognitive Level: Higher _

Lower _X

Question Source: New X_

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

4.1 General Requirements for Tagging (continued)

12. Responsibility may be transferred or assumed by another Clearance Owner.

A. Responsibility as the Clearance Owner SHALL be transferred or assumed by another Clearance Owner when they sign onto the Clearance.

B. Operations Shift Supervision may sign off another Clearance Owner after making reasonable attempts to notify them. The individual that was "signed for" shall be informed upon return to work in accordance with instructions provided in this document.

13. In the case of eSOMS system failure, refer to OP-AA-101-1000-F02 Paper Based Tagging Instructions until eSOMS system is restored to service.

14. It may not always be possible to de-energize all equipment or equipment components within the requested boundary. When such a situation arises, Operations Shift Supervision shall ensure that the Clearance Owner fully understands what equipment is energized and where it is located within the Clearance boundary.

15. Status Control Clearances that are developed and hung solely to maintain plant Status Control:

A. Should reside in a Status Control Clearance Group folder in eSOMS.

B. Shall not be used to directly support maintenance activities; may be utilized when maintenance activities require operational Status Control not specifically required by the maintenance activity.

C. Shall not use Danger Tags.

D. Attachments associated with the request, preparation, review, implementation, etc. of Status Control Clearances may be waived by Operations Shift Supervision.

E. Shall be accepted by Operations Shift Supervision as a Clearance Owner.

Question # 69

Given the following:

- Unit 4 just completed a reactor startup and reached 2% power.

Subsequently:

- 4A Containment Spray pump is declared INOPERABLE.
- Site management decided that the repair will take 5 days and that power will be maintained at 1-2% until the pump is restored.

Which one of the following completes the statements below?

IAW OP-AA-1000**,** Conduct Of Infrequently Performed Tests Or Evolutions (IPTE), the crew will receive an IPTE brief to discuss __(1)__.

The IPTE Manager __(2)__ required to hold an active SRO license.

A. (1) contingency actions for degraded containment depressurization capability
(2) is

B. (1) extended operations at low power
(2) is

C. (1) contingency actions for degraded containment depressurization capability
(2) is NOT

D. (1) extended operations at low power
(2) is NOT

Answer Analysis

Discussion: Correct answer is D. IPTEs are required for extended low power operation below 10% power. For operations department related evolutions or tests, a senior level manager with a current or former license is required if acting for the IPTE manager in IPTE related activities.

A. Incorrect..
   Part 1 incorrect, plausible since contingency actions for degraded systems should be briefed, but this is not an IPTE.
   Part 2 incorrect, plausible since an SRO with an active license is allowed to serve as IPTE manager.

B. Incorrect.
   Part 1 correct.
   Part 2 incorrect, plausible since an SRO with an active license is allowed to serve as IPTE manager.

C. Incorrect.
   Part 1 incorrect, plausible since contingency actions for degraded systems should be briefed, but this is not an IPTE.
   Part 2 correct.

D. Correct, see discussion above

Question Number: 69

Tier: 3 **Category 2: Equipment Control**

K/A: **G2.2.7 Knowledge of the process for conducting special or infrequent tests.**

Importance Rating: **2.9** **10 CFR Part 55:** 41.10

10 CFR 55.43.b **:**N/A

K/A Match: **RO Question. (Tier 3, Conduct of Ops)** The K/A is matched as the applicant must evaluate why an IPTE is required. Also the applicant must determine if the process allows formerly licensed individuals to act for an Operations IPTE manager.

SRO Justification: N/A

Technical References: **OP-AA-1000,** CONDUCT OF INFREQUENTLY PERFORMED TESTS OR EVOLUTIONS

Proposed references to be provided: None

Learning Objective: PTN 6902045 OBJ. 2

Cognitive Level: Higher _

Lower x_

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

2.0 TERMS AND DEFINITIONS (continued)

2. **IPTE Manager:** A member of the plant staff or other individual designated by the Site Director, to provide oversight of a specific evolution. Specific required attributes include:

   - A senior level member of management (e.g., in a role senior to the Shift Manager).
   - Is **NOT** involved in performance of the assigned evolution. Specifically, this individual does not replace any individual involved in the test or evolution nor supervises the evolution.
   - Shall possess the requisite knowledge, skills and experience to provide meaningful oversight of the evolution.
   - Shall provide appropriate oversight based on the complexity of the test or evolution.
   - Must hold, or have held an SRO license or equivalent certification if acting as the IPTE Manager for Operations led activities.

   Depending on the complexity and risk of the test or evolution being conducted, the role of the IPTE Manager may be reduced to frequent observation or spot checking with the approval of the Site Director and the Director of Operations.

3. **Lead Test Coordinator:** An individual assigned by the Section Manager responsible for the evolution to supervise the entire test/evolution. This could be anyone with an in-depth knowledge of the test/evolution. Lead Test Coordinator is meant to be a generic title; it could be the Test Director, SVI Coordinator, etc. The Lead Test Coordinator shall **NOT** be considered an “oversight” role.

ATTACHMENT 1
IPTE EXAMPLES
(Page 1 of 1)

Listed below are activities that are classified as Infrequently Performed Tests or Evolutions (IPTEs) based on the coordination or complexity AND the need for additional management oversight:

- Reactor Startups
- Planned Reactor Shutdowns
- Reactor Physics Testing Procedures
- Integrated Safety System Testing
- Containment Integrated Leak Rate Test
- Turbine Overspeed Testing
- Drain and Fill of the Reactor Coolant System
- Operation or intent to operate, at Lowered/Reduced inventory
- Reactor Head Lift and Installation, including core internals
- Steam Generator Nozzle Dam Installation/Removal
- Use of a Freeze Seal
- Pressure Vessel Class I leak tests
- Special Test Procedure
- Conditions which could cause an inadvertent dilution (PWR's only)
- Extended Operations at low power (less than 10%)
- Control Rod Sequence changes (DAEC)
- Operations with potential to drain the RPV (OPDRVs)
- Movement of irradiated fuel, including Core Alterations
- Other activities designated in Station procedures as IPTE

Question # 70

Which one of the following completes the statements below?

If the 3A EDG Starting Air Compressor failed to start 3-ONOP-023, Emergency Diesel Generator Failure, will direct to ______.

A. cross-tie EDGs starting air compressors

B. start the diesel air compressor

C. align Nitrogen Cylinder Backup

D. install a temporary Starting Air Compressor

Answer Analysis

A. Correct. The ONOP action for low start without an air compressor is to cross tie to Air Start Systems.

B. Incorrect, Plausible since this would be the correct action for Unit 4 EDGs.

C. Incorrect, plausible since N2 backup is used for Unit 3 EDG for other reasons, i.e. EDG day tank fill when no instrument air is present.

D. Incorrect, plausible since other plant procedures contain direction to install temporary compressors, i.e. 0-NOP-013.02, Temporary Instrument Air Compressor Operation.

Question Number: 70

Tier: 3 **Group:**

K/A: Equipment Control; 2.2.4, (multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.

Importance Rating: 3.6

10 CFR Part 55: 41.6, 41.7, 41.10

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-ONOP-023.2

Proposed references to be provided: None

Learning Objective: PTN 6902270 OBJ. 8

Cognitive Level: Higher _

	Lower	_X
Question Source:	New	x
	Modified Bank	_
	Bank	_
Question History:	New	
Comments:	Screenshot of development references as applicable	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	**Place Affected Emergency Diesel Generator Master Control Switch In LOCAL**	
3	**Check Low Start Air Pressure Light – OFF**	Perform the following: a. **IF** starting air system low pressure is caused by leakage from the system, **THEN** identify and isolate the leak. b. Verify affected EDGs starting air compressor is running. c. **IF** affected EDGs starting air compressor can **NOT** be started, **THEN** cross-tie EDGs starting air compressors using 3-OP-023, EMERGENCY DIESEL GENERATOR.
4	**Check Control Power Light - ON**	Request RO to verify affected emergency Diesel Generator Flashing and Control Power breaker - ON. • 3D01-47 for 3A EDG • 3D23A-28 for 3B EDG
5	**Check Skid Tank Level Low Light-OFF**	Perform the following: a. Proceed to affected EDG Day Tank Room. b. Verify Day Tank level - BETWEEN 4 FT-9 INCHES AND 6 FT - 2 INCHES. c. Verify Diesel Oil Day Tank Outlet to EDG Skid Tank Isolation Valve – OPEN. • 3-70-026A for EDG 3A • 3-70-026B for EDG 3B d. Proceed to affected EDG Room. e. Gravity feed fuel using one of the following: 1) PUSH-TO-FILL pushbutton on northwest side of EDG. **OR** 2) Skid Tank Solenoid Valve Bypass Line Isolation on southwest side of EDG. • 3-70-048A for EDG 3A • 3-70-048B for EDG 3B

Question # 71

Given the following:

- Unit 4 is at 100% power.
- ANN X4/1, ARMS HI RADIATION, alarms

Which one of the following completes the statements below?

The RO will check the alarming channel by inspection of a panel on the __(1)__ side of the Control Room?
IAW 0-ONOP-066, ARMS High Radiation, once the alarming channel is identified the RO will NEXT press the __(2)__.

A. (1) Unit 4
(2) ALARM ACK pushbutton to silence the alarm

B. (1) Unit 4
(2) HIGH Alarm pushbutton to read alarm setpoint

C. (1) Unit 3
(2) ALARM ACK pushbutton to silence the alarm

D. (1) Unit 3
(2) HIGH Alarm pushbutton to read alarm setpoint

Answer Analysis

A. Incorrect.
Part 1 incorrect, plausible since plausible since there are other instrumentation cabinets that are unique to the Unit 4 side of the Control Room, i.e. 4QR81 and 4QR82 Instrumentation Racks which contain all the Control Room Ventilation Controls.
Part 2 incorrect, plausible since this is an action in 3-ONOP-067.

B. Incorrect.
Part 1 incorrect, plausible since plausible since there are other instrumentation cabinets that are unique to the Unit 4 side of the Control Room, i.e. 4QR81 and 4QR82 Instrumentation Racks which contain all the Control Room Ventilation Controls.
Part 2 correct.

C. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since this is an action in 3-ONOP-067.

D. Correct. The indication for the ARMs channels is located in the cabinet on the Unit 3 side of the control room. IAW 3-ONOP-067, the setpoint for the ARM channel is verified FIRST.

Question Number: 71

Tier: 3 **Category 3: Radiation Control**

K/A: **G2.3.5 Ability to use Radiation Monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc**

Importance Rating: **2.9; 10 CFR Part 55:** 41.11,12

10 CFR 55.43.b **:**N/A

K/A Match: The K/A is matched as the applicant must evaluate plant conditions, identify the proper ONOP and sequence of actions to respond to **(USE)** a R-30 Rad Monitor alarm to lessen the impact on personnel.

SRO Justification: N/A

Technical References:

ARP X 4/1, ARMS High Radiation

0-ONOP – 066, High Area Radiation Monitoring System Alarm

0-ADM-200

Proposed references to be provided: None

Learning Objective: PTN 6902242 OBJ. 6

Cognitive Level: Higher _

Lower x_

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1 **Identify Alarming ARMS Channel(s)**

Identify individual channel(s) HIGH alarm light illuminated on ARMS control panel R-30

NOTE

The alarm should not be silenced until personnel have been evacuated from the affected area and entry to the area has been restricted OR the alarm has been determined to be an invalid alarm.

2 **Verify Alarm Is Valid**

a. Press HIGH alarm pushbutton on ARMS control panel to read alarm setpoint

b. Digital display matched high alarm setpoint on Attachment 1

b. Perform the following:

1. Press the ALARM ACK pushbutton on ARMS control panel R-30 to silence the alarm.
2. Announce that access has been restored to the evacuated area.
3. Notify RP to install a portable area radiation monitor which meets 10CFR50.68(b) requirements if affected channel is 21 or higher.
4. **IF** any ARM channel 1 through 6 fails while in Mode 5 or 6, **THEN** notify Radiation Protection to install a portable area radiation monitor with alarm
5. Initiate a PWO to repair the defective ARMS channel.
6. Return to procedure and step in effect.

CAUTION

If special fire protection requirements (fire watch etc.) are required for the building or area being evacuated, the fire protection requirements should be re-established as soon as plant conditions allow.

3 **Make Plant Announcement**

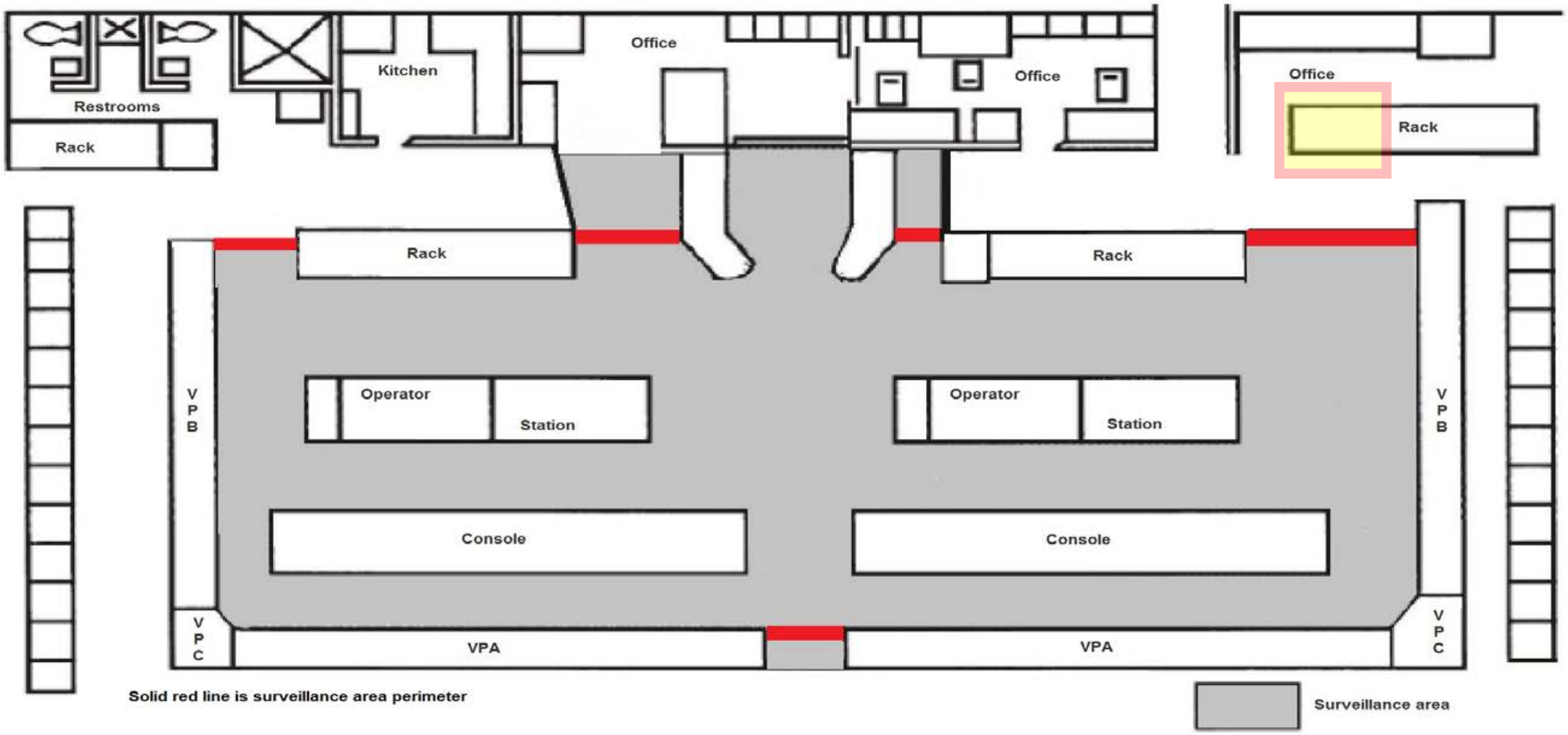
a. Use the plant page system with page boost

b. Announce the affected areas

c. Notify people to keep clear

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Check Alarming Channel NOT Inside Containment**	**IF** people are inside containment, **THEN** sound Containment Evacuation Alarm for affected unit.
5	**Notify RP To Survey the Area To Determine The Source Of Radiation**	
	a. RP survey of affected area indicates an increased radiation level b. Coordinate with the RPSS on possible evacuation routes and changing radiological conditions.	a. Perform the following: 1. Press the ALARM ACK pushbutton on ARMS control panel R-30 to silence the alarm. 2. Announce that access has been restored to the evacuated area. 3. Discontinue area survey. 4. Notify RP to install a portable area radiation monitor which meets 10CFR50.68(b) requirements if affected channel is 21 or higher. 5. **IF** any ARM channel 1 through 6 fails while in Mode 5 or 6, **THEN** notify Radiation Protection to install a portable area radiation monitor with alarm 6. Initiate a PWO to repair the defective ARMS channel. 7. Return to procedure and step in effect.
6	**Check ONLY One ARM Channel For The Affected Area Indicates Increased Radiation Levels**	Perform the following: a. Evacuate the area. b. Notify RP of increased radiation levels on ARMS channels. c. Notify Security to restrict entry to the affected area(s).
7	**Press ALARM ACK Pushbutton On ARMS Control Panel**	

ATTACHMENT 1
Surveillance Area
(Page 1 of 1)



The surveillance area boundaries are marked in red on the Control Room floor for clarity. The limited areas between the nuclear instrumentation/process radiation monitor racks and the Control Room boundary wall are **NOT** authorized for extended occupancy. Permission is required from RCO, CRS, or SM to access into the surveillance area.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	072		A4.01
	Importance Rating	3.0		

Ability to manually operate and/or monitor in the control room: Alarm and interlock setpoint checks and adjustments

Proposed Question: RO Question # 64

Given the following conditions:

- Unit 3 is in MODE 5.
- ANN X4/1, ARMS HI RADIATION, alarms.

Which one of the following completes the statements below?

0-ONOP-066, High Area Radiation Monitoring System Alarm, requires the crew to FIRST __(1)__.

These actions will be performed from a panel located __(2)__ the RO surveillance area.

A.	(1) verify the setpoint on the affected ARMS channel, AND then press the alarm acknowledge pushbutton (2) inside
B.	(1) press the alarm acknowledge pushbutton, AND then verify the setpoint on the affected ARMS (2) inside
C.	(1) verify the setpoint on the affected ARMS channel, AND then press the alarm acknowledge pushbutton (2) outside
D.	(1) press the alarm acknowledge pushbutton, AND then verify the setpoint on the affected ARMS (2) outside

Proposed Answer:	C

A.	**Incorrect** Part 2 incorrect, plausible since the PRMS cabinet is located in the surveillance area.
B.	**Incorrect** Part 1 incorrect; plausible since later in the procedure the alarming channel alarm is acknowledged. 0-ONOP-066 contains a caution that alerts the operator not to silence the alarm until the area has been evacuated or the alarm has been determined to be invalid.
C.	**Correct** IAW 0-ONOP-066.
D.	**Incorrect** Both parts are incorrect; plausible per discussion above. Question parts are independent and plausible as a whole.

Technical Reference(s)	0-ONOP-066		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902246 Obj. 6		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	11
	55.43	
Purpose and operation of radiation monitoring systems, including alarms and survey equipment.		

Question # 72

Which one of the following completes the statements below?

IAW RP-AA-103-1002, High Radiation Area Controls, in order to access a HIGH Radiation Area, and operator will sign onto an RWP with a dose rate alarm setpoint that is __(1)__than 100mr/hr; a briefing with RP __(2)__ required.

A. (1) greater
(2) is

B. (1) greater
(2) is NOT

C. (1) less
(2) is

D. (1) less
(2) is NOT

Answer Analysis

A. Correct. IAW RP-AA-103-1002, a High Radiation Area is defined as an area with 0.1 Rem in one hour at 30 cm. A brief with RP is required before permission is granted.

B. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since entry into radiation areas does not require a brief with RP.

C. Incorrect.
Part 1 incorrect, plausible since 100 mr/hr is the lower limit for the High radiation area. There could be a misconception that this limit is not to be challenged.
Part 2 correct.

D. Incorrect.
Part 1 incorrect, plausible since 100 mr/hr is the lower limit for the High radiation area. There could be a misconception that this limit is not to be challenged.
Part 2 incorrect, plausible since entry into radiation areas does not require a brief with RP.

Question Number: 72

Tier: 3 **Group:**

K/A: Radiation Control; 2.3.7, Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Importance Rating: 3.5

10 CFR Part 55: 41.12

10 CFR 55.43.b : N/A

K/A Match:

SRO Justification: N/A

Technical References: RP-AA-103-1002

Proposed references to be provided: None

Learning Objective: 6902970 Objective 2

Cognitive Level: Higher _

Lower X_

Question Source: New _X

Modified Bank _

Bank

Question History: New

Comments: Screenshot of development references as applicable

2.0 TERMS AND DEFINITIONS (continued)

7. Double Verification - This consists of a second individual performing a peer-check to insure that the door is locked/secure.

8. Entrance or Access Point - Any location through which an individual could gain access to radiation areas. This includes accessible entry or exit portals large enough to permit human entry, irrespective of design use.

9. Guarded or Guarding - Actions taken or implemented to protect or prevent inadvertent or unauthorized entry. This includes maintaining continuous awareness of personnel approaching an entry point.

10. High Radiation Area (HRA) - Any area accessible to individuals in which radiation levels from a radiation source external to the body could result in an individual receiving a dose equivalent in excess of 0.1 Rem (Deep Dose Equivalent) in 1 hour at 30 cm (11.8 in) from the radiation source or from any surface that the radiation penetrates.

11. Key Accountability - maintaining issued RP High Radiation Area Key in personal possession at all times.

12. Key Holder - an RP qualified person who maintains possession, control, and issuance of keys from the Locked High Radiation Area Key Locker.

13. Key User – an RP qualified person issued a LHRA or VHRA key, who maintains possession and control of the key until returned to the Key Holder.

14. Lock - A device for securing a door, gate, or the like in position when closed, consisting of a bolt or system of bolts propelled and withdrawn by a core operated by a key.

15. Lock Check/Door Check - Physical inspection of a locking mechanism to ensure the lock meets the intended purpose of maintaining the door or gate in a fixed position, usually closed.

16. Lock Core/Core - Inner mechanism which controls opening or closing the lock with a key.

17. Lockset/Core Set - Primarily the lock core and the key which operates the lock when the core is installed.

    - Locksets could be the entire assembly of parts making up a complete locking system, including knobs, plates, and a lock mechanism.

4.2 RP Requirements to Allow Access to High Radiation Areas (continued)

4. **BRIEF** workers using RP-AA-103-1002-F02, HRA / LHRA /VHRA Briefing Acknowledgement Form.

5. **PERFORM** the following when conducting briefings:

   A. **ENSURE** RWP is in hand.

   B. **IDENTIFY** purpose. (examples: "This is a High Radiation Area brief", "This concludes your High Radiation Area brief", etc.)

> **NOTE**
>
> Radiation Protection Personnel qualified in Radiation Protection procedures (applicable to Technical Specification, High Radiation Area) are not required to have a briefing documented on RP-AA-103-1002-F02 form.

6. **ENSURE** personnel requesting HRA access are on correct RWP and Task using Sentinel or telemetry software.

7. **DETERMINE** if additional exposure control barriers are needed by referring to the flowchart in Attachment 1, HRA / LHRA Area Extra Barrier Logic.

8. **FORWARD** completed RP-AA-103-1002-F02 form to RP Supervisor.

9. Form RP-AA-103-1002-F02 does not meet the requirements of a quality document in accordance with RM-AA-100 RECORDS MANAGEMENT PROGRAM and is not required to be transmitted to QA Records for retention.

Question # 73

Which one of the following completes the statements below?

IAW 0-ADM-211, Emergency and OFF-Normal Operating Procedure Usage, ONOP actions that must be performed from memory are defined as __(1)__.

IAW 0-ADM-211, entry into __(2)__ unless specifically exempted.

A. (1) Immediate Operator Actions
(2) ONOPs takes precedence over ARP actions

B. (1) Immediate Operator Actions
(2) ARPs takes precedence over ONOP actions

C. (1) Prompt Actions
(2) ONOPs takes precedence over ARP actions

D. (1) Prompt Actions
(2) ARPs takes precedence over ONOP actions

Answer Analysis

A. Correct. ONOP actions performed from memory are Immediate Operator Actions. ONOPs are higher priority procedures than ARPs

B. Incorrect.
Part 1 correct.
Part 2 incorrect, plausible since, normally when an alarm comes in and depending on the scenario, the operator NORMALLY reviews the ARP first.

C. Incorrect.
Part 1 incorrect, plausible since ARP have Prompt Actions, which are performed in a similar way to IOAs.
Part 2 correct.

D. Incorrect.
Part 1 incorrect, plausible since ARP have Prompt Actions, which are performed in a similar way to IOAs.
Part 2 incorrect, plausible since, normally when an alarm comes in and depending on the scenario, the operator NORMALLY reviews the ARP first.

Question Number: 73

Tier: 3 **Category 3: Emergency Procedures/Plan**

K/A: **G2.4.11 Knowledge of Abnormal Condition Procedures**

Importance Rating: **3.4; - 10 CFR Part 55:** 41.10

10 CFR 55.43.b :N/A

K/A Match: The K/A is matched as the applicant must understand format and rules of use for ARPs and ONOPs.

SRO Justification: N/A

Technical References: 0-ADM-211 Emergency and Off Normal Procedure Usage

Proposed references to be provided: None

Learning Objective: PTN 6902320 OBJ. 2

Cognitive Level: Higher _

Lower x_

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

2.0 TERMS AND DEFINITIONS (continued)

5. Faulted Steam Generator

Refers to any steam generator with an unisolable leak in its secondary pressure boundary of sufficient size to require Safety Injection. Indications of a faulted SG is one that is completely depressurized or one that has SG pressure decreasing in an uncontrolled manner (See Definition Uncontrolled)

6. Functional Restoration Procedures (FRPs)

Those procedures that respond to Critical Safety Function challenges. Guidance is provided to restore the Critical Safety Function to a satisfied condition. Typically, actions are based on the severity of the challenge and may **NOT** correspond to good operational practice.

7. Immediate/Immediately

The definition of immediate should be based on the specific situation. The action should be performed in a timely manner. In some cases, this may require action in less than 15 minutes and in others, 1 hour may be appropriate. (This definition is **NOT** for Immediate Operator Actions.)

8. Immediate Operator Action (IOA)

Time sensitive tasks that are of such high importance that they must be performed from memory to ensure that they performed in an expeditious manner. IOA steps are generally reserved for the highest priority steps that must be performed to ensure and maintain reactor safety.

9. Local (Locally)

An action performed by an operator outside the Control Room.

10. Manual (Manually)

An action performed by the operator in the Control Room. This does **NOT** include automatic actions, which take place without operator intervention.

4.2 Procedure Hierarchy

1. Entry into the EOPs takes precedence over actions specified in other plant procedures unless specifically exempted.

2. Within the EOPs, entry into Red and Orange path FRPs take precedence over actions specified in E-series EOPs unless specifically exempted.

3. Entry into ONOPs (AOPs) takes precedence over action specified in ARPs unless specifically exempted.

4. Prompt actions from ARP should be read prior to ONOP (AOP) entry.

4.3 Command and Control

1. The Control Room Team must avoid a false sense of either urgency or overconfidence during event mitigation. The pace of implementation will be controlled by the Procedure Director and monitored by the SM.

2. Commands to the panel operators will come from the Procedure Director. The SM should direct any commands through the Procedure Director.

3. Communication from the panel operators should normally be directed to the Procedure Director.

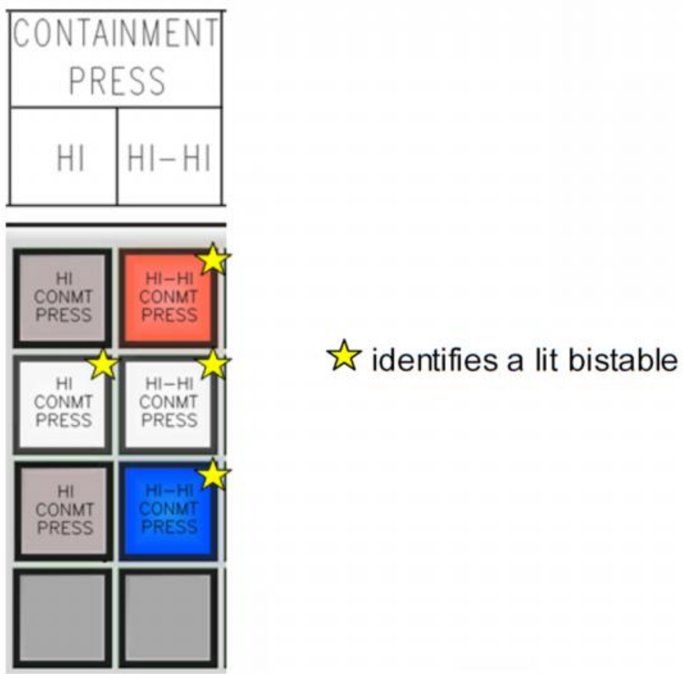
4. Communications from the STA, associated with CSFST transitions, should be directed to the Procedure Director. Communications from the STA, associated with the procedure oversight, should normally be directed to the SM who will share them with the Procedure Director or Control Room Team as appropriate.

   A. When the STA has pertinent information that is important to the Control Room Team, the STA should perform a crew update to communicate the information to the team.

Question # 74

Given the following:

- Unit 3 trips from full power due to a steam line break inside containment.
- Containment pressure on DCS is 22 psig.
- The following is observed on VPB:



ALSO PROVIDED AS A REFERENCE IN LARGER FORMAT

Which one of the following completes the statements below? Assume NO operator action.

Assuming no operator action, Containment spray pumps __(1)__ AUTOMATICALLY started. An ORANGE path on the Containment critical safety function, __(2)__ present.

A. (1) have
(2) is

B. (1) have NOT
(2) is

C. (1) have
(2) is NOT

D. (1) have NOT

(2) is NOT

Answer Analysis

Discussion of system operation and conditions that relates to the correct answer choice.

A. (1) Incorrect, Plausible if candidate believes only HI HI containment pressure is required to actuate containment spray.
(2) Correct

B. Correct. 2/3 HI and HI HI containment pressure signals are required to start containment spray pumps. In this case only 2/3 HI HI logic is made up therefore pumps will NOT have started. Orange path on Containment critical safety function is present with containment pressure > 20 psig and no containment spray pumps running.

C. (1) Incorrect, see A(1)
(2) Incorrect, Plausible if candidate believes containment spray pumps have started therefore FRP entry is NOT required.

D. (1) Correct
(2) Incorrect, This combination is plausible if the candidate believes pumps have not started and confuses ORANGE path FR-Z.1 entry with RED path entry conditions. Candidate thinks the RED path condition applies.

Question Number: 74

Tier: 3 **Group:**

K/A: Emergency Procedures/Plan; 2.4.21, Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Importance Rating: 4.0

10 CFR Part 55: 41.7

10 CFR 55.43.b : N/A

K/A Match: The K/A is matched as the applicant

SRO Justification: N/A

Technical References: 3-EOP-F-0, 5610-T-L1 SH 11

Proposed references to be provided: None

Learning Objective: PTN 6902922 OBJ. 15

Cognitive Level: Higher X_

Lower _

Question Source: New _

Modified Bank _

Bank X

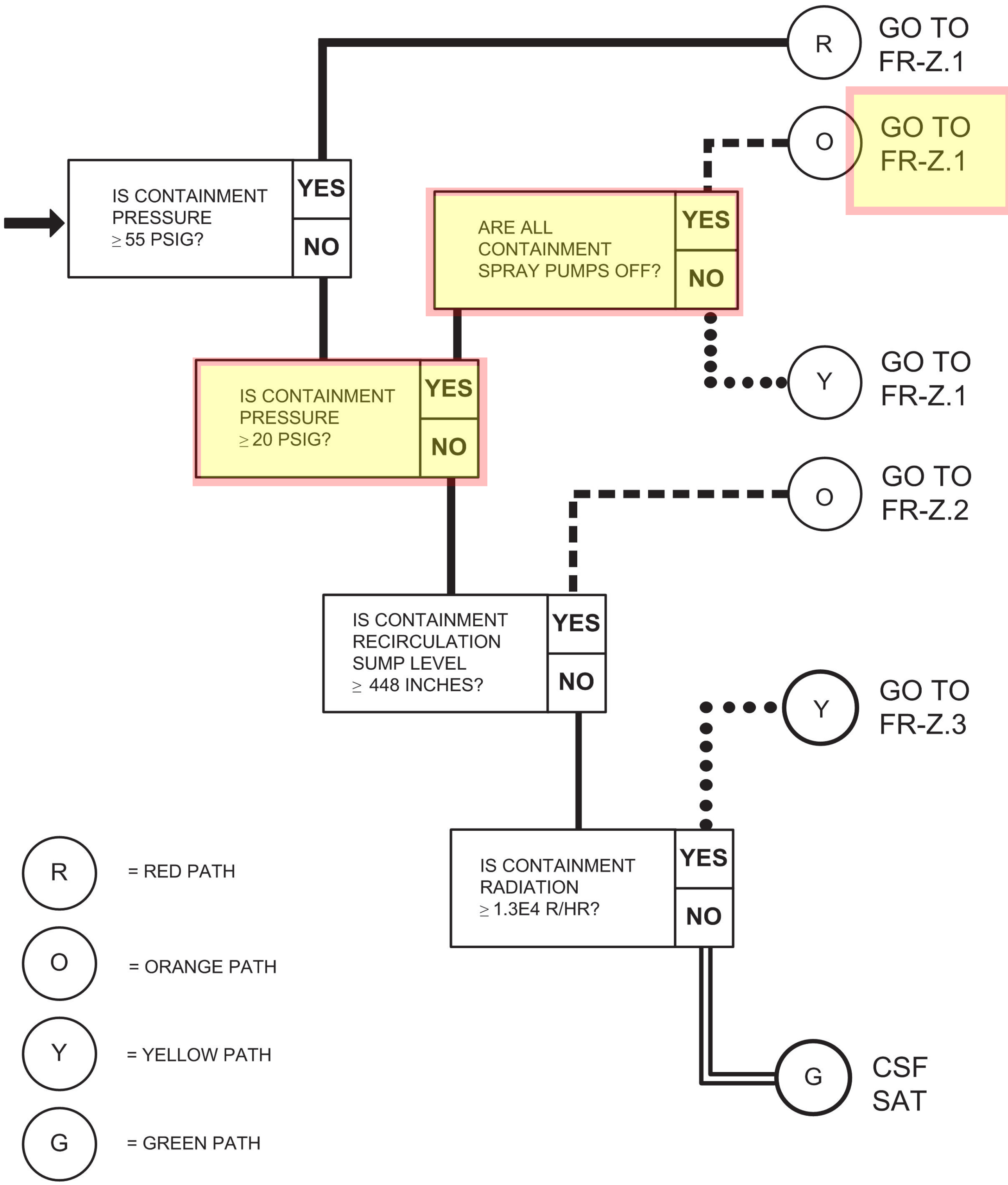
Question History: Question used as new question on the 2016 PTN NRC exam as RO question # 40

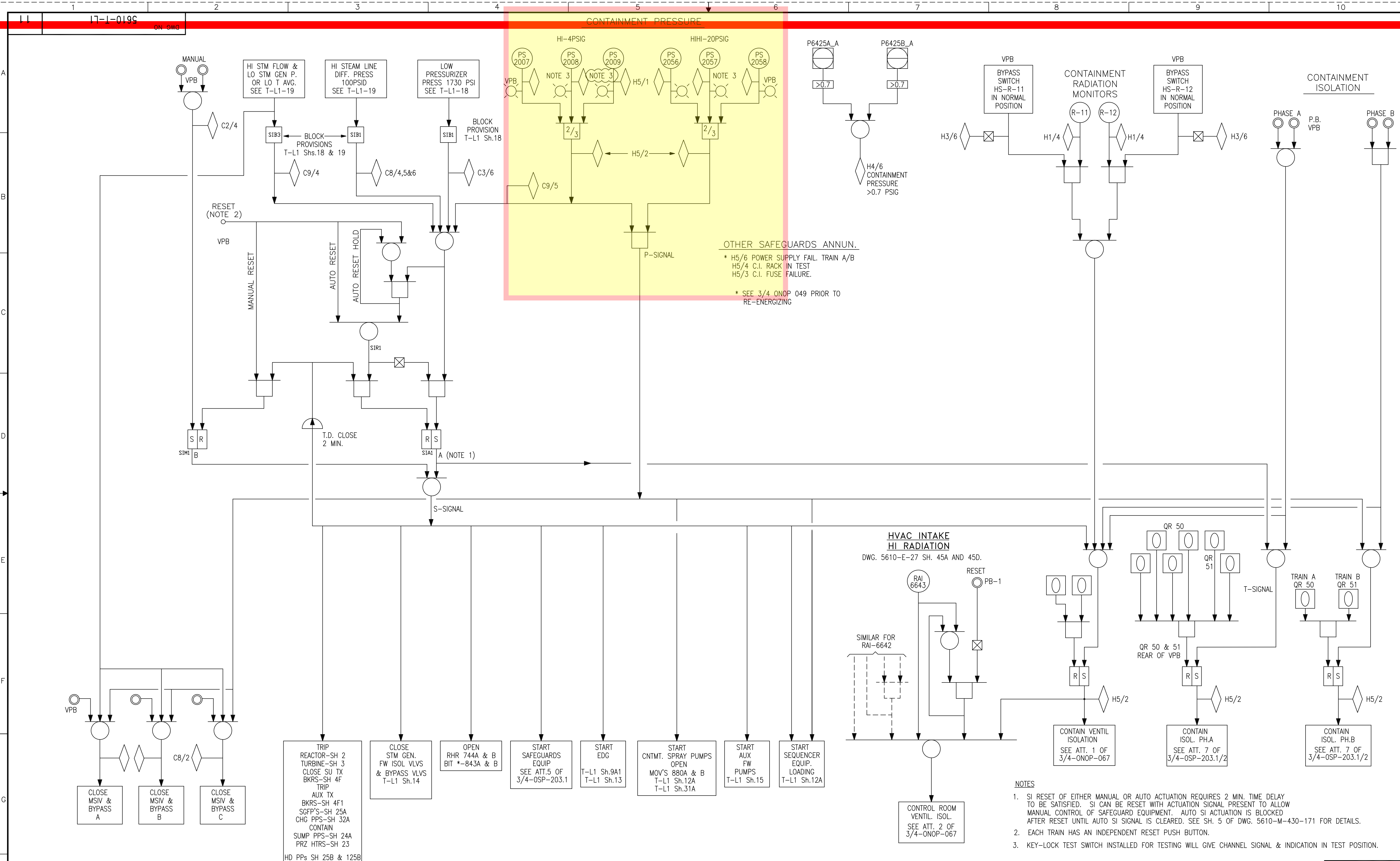
Comments: Screenshot of development references as applicable

ENCLOSURE 5

CSF F-0.5 Containment

(Page 1 of 1)





NOTES

1. SI RESET OF EITHER MANUAL OR AUTO ACTUATION REQUIRES 2 MIN. TIME DELAY TO BE SATISFIED. SI CAN BE RESET WITH ACTUATION SIGNAL PRESENT TO ALLOW MANUAL CONTROL OF SAFEGUARD EQUIPMENT. AUTO SI ACTUATION IS BLOCKED AFTER RESET UNTIL AUTO SI SIGNAL IS CLEARED. SEE SH. 5 OF DWG. 5610-M-430-171 FOR DETAILS.
2. EACH TRAIN HAS AN INDEPENDENT RESET PUSH BUTTON.
3. KEY-LOCK TEST SWITCH INSTALLED FOR TESTING WILL GIVE CHANNEL SIGNAL & INDICATION IN TEST POSITION.

SAFETY RELATED | POD

REV	DATE	REVISION	BY	CH	APP	APP
33	10-19-17	ISSUED AS-BUILT PER EC 289667.	RH	CW	–	BB
32	2-22-16	ISSUED AS-BUILT PER EC-DCR 285837.	RH	AFG	–	SB
31	08-28-12	ISSUED AS-BUILT PER EC 246935 (PC/M 08-187).	RV	RH	–	PMB
30	12-28-09	ISSUED AS-BUILT PER CRN-E-18083 (PC/M 09-106).	RV	BB	–	LG
29	11-20-09	ISSUED AS-BUILT PER PC/M 08-037.	RV	RH	–	AD
28	4-9-09	ISSUED AS-BUILT PER PC/M 08-006.	RH	KCP	–	RCS

REV	DATE	REVISION	BY	CH	APP	APP
27	07-15-03	ISSUED AS-BUILT PER CRN-E-16419 (PC/M 02-065).	RV	RH	BP	TS
26	1-20-98	ISSUED AS-BUILT PER CRN'S I-3659 & I-3663(PC/M 97-042).	RV	RH	MA	TS
25	11-7-97	ISSUED AS-BUILT PER CRN-I-3640(PC/M 97-036)	RV	RH	MA	TS/JG
24	1-23-97	ISSUED AS-BUILT PER CRN-I-3482(PC/M 96-080)	RV	RH	BP	TS
23	9-30-96	INCORPORATED CRN-I-3418 FOR PC/M 96-058.	RV	BP	–	TS
34	–	ISSUED AS-BUILT PER EC 291574.	SEE DUP FOR INITIALS & DATE			

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

FLORIDA POWER & LIGHT

SAFEGUARDS ACTUATION AND STEAM LINE ISOLATION

POD

DRAWING NUMBER 5610-T-L1

SHEET 11

SYS 063

REV 34

5610-T-L1_SH 11.DWG

Question # 75

Given the following:

- Unit 3 experiences a 3A SGTR from 100% power.

Which one of the following completes the statements below?

Once the first two Major Action Categories have been completed to isolate the 3A steam generator and establish subcooling margin, the third major action depressurizes RCS to:

A. prevent a pressurized thermal shock of the RCS

B. Refill the pressurizer

C. Allow placing RHR in service

D. Reduce RCS pressure to below shutoff head of the HHSI pumps

Answer Analysis

Discussion: There are 5 Major Action Categories in E-3

Identify and Isolate Ruptured SG(s)

Cool Down to Establish RCS Subcooling Margin

Depressurize RCS to Restore Inventory

Terminate SI to Stop Primary-to-Secondary Leakage

Prepare for Cooldown to Cold Shutdown

The question addresses item 3 to restore inventory by refilling the pressurizer.

A. Incorrect: It is plausible that lowering RCS pressure aids in preventing PTS of the RCS. This is incorrect since this is not the basis of the 3-EOP-E-3 step.

B. Correct

C. Incorrect. It is plausible that depressurizing to place RHR in service might be correct as RHR shutoff pressure head is below that of the RCS and the steam generator.

D. Incorrect. Reducing RCS pressure to below the shutoff head of the HHSI pumps would allow those pumps to inject and could be seen as plausible. However RCS pressure would still possibly (likely) be above the 3A steam generator pressure and would not achieve the goal to refill the pressurizer for the upcoming RCS cooldown.

Question Number: 75

Tier: 3 **Category 4: Emergency Procedures/Plan**

K/A: **G2.4.78 Knowledge of event based EOP mitigation strategies**

Importance Rating: **3.1; 10 CFR Part 55:** 41.10

10 CFR 55.43.b :N/A

K/A Match: **RO question:** The K/A is matched as the applicant must have knowledge of the Major Action Categories, i.e. the overall mitigation strategy to combat a SGTR.

SRO Justification: N/A

Technical References: E-3 Steam Generator Tube Rupture Basis Document

Proposed references to be provided: None

Learning Objective: PTN 6902339 OBJ. 4

Cognitive Level: Higher _

Lower x_

Question Source: New X

Modified Bank _

Bank _

Question History: New

Comments: Screenshot of development references as applicable

BD-EOP-E-3

Steam Generator Tube Rupture

Depressurize RCS to Restore Inventory

After the cooldown is completed, the system response is monitored to verify proper isolation of the

affected steam generator and correct event diagnosis. At this time, pressure in the affected steam

generators should be greater than the pressure in the intact steam generators and increasing slowly.

The RCS should also be subcooled. If these symptoms are not evident, the operator is directed to

ECA-3.1 to continue RCS cooldown and depressurization and to reduce SI flow while ensuring

adequate coolant inventory.

When correct event diagnosis has been verified, recovery actions continue in E-3 by depressurizing

the RCS to minimize primary-to-secondary leakage and to refill the pressurizer. If RCPs are

running, maximum normal spray is initiated. A pressurizer PORV or auxiliary spray can also be

used if normal spray is not available, such as following a station blackout. Contingency actions are

also presented in ECA-3.3 for terminating SI flow without pressurizer level in the unlikely case

that no pressurizer pressure control is available.

RCS pressure is decreased until pressurizer level has been restored and RCS pressure is equal to (or

close to with the alternate criteria) the affected steam generator pressure. During natural

circulation conditions, the upper head region could void. Since this would cause pressurizer level

to increase rapidly, RCS depressurization is stopped on high level in the pressurizer to avoid a

water solid condition. RCS depressurization is also stopped if pressure approaches saturation to

avoid steam formation in the active RCS regions.

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Prepare for Cooldown to Cold Shutdown

The operator must also select the best post-SGTR cooldown method based on an evaluation of the

plant status.

When the instructions provided in E-3 are completed, primary-to-secondary leakage and

radiological releases from the affected steam generator(s) should be stopped. However, the plant should be cooled and depressurized to cold shutdown conditions in order to make repairs and

ensure no further radiological releases. This cooldown and depressurization is complicated by the

ruptured steam generator which will act like a second pressurizer and inhibit RCS depressurization.

Three alternate means of performing the post-SGTR cooldown have been developed and are

presented in ES-3.1, POST-SGTR COOLDOWN USING BACKFILL

BASIS DOCUMENT

Cool Down to Establish RCS Subcooling Margin

Isolation of the affected steam generators is followed by a rapid cooldown of the RCS using the intact steam generators. The target temperature is determined from the pressure in the affected steam generator since RCS pressure must be reduced to this value to stop primary-to-secondary leakage. The core exit fluid temperature must be decreased sufficiently to ensure subcooling at the affected steam generator pressure.

The preferred means for cooling the RCS is steam dump to condenser. This minimizes radiological releases, conserves feedwater, and provides the most rapid cooldown capability. Atmospheric relief valves on the intact steam generators can also be used if steam dump to condenser is not possible. In the unlikely event that no intact steam generator is available, a faulted or ruptured steam generator must be used to cool the RCS. If a ruptured steam generator is used, the operator is directed to ECA-3.1 to limit leakage from the RCS until RHR System cooling can be established.

If necessary, charging pumps are started during the cooldown to restore RCP seal injection flow and to aid in refilling the pressurizer during the RCS depressurization in the following steps. When the required core exit fluid temperature is reached, the cooldown is stopped and steam flow is controlled to stabilize temperatures.

Depressurize RCS to Restore Inventory

After the cooldown is completed, the system response is monitored to verify proper isolation of the affected steam generator and correct event diagnosis. At this time, pressure in the affected steam generators should be greater than the pressure in the intact steam generators and increasing slowly. The RCS should also be subcooled. If these symptoms are not evident, the operator is directed to ECA-3.1 to continue RCS cooldown and depressurization and to reduce SI flow while ensuring adequate coolant inventory.

When correct event diagnosis has been verified, recovery actions continue in E-3 by depressurizing the RCS to minimize primary-to-secondary leakage and to refill the pressurizer. If RCPs are running, maximum normal spray is initiated. A pressurizer PORV or auxiliary spray can also be used if normal spray is not available, such as following a station blackout. Contingency actions are also presented in ECA-3.3 for terminating SI flow without pressurizer level in the unlikely case that no pressurizer pressure control is available.

RCS pressure is decreased until pressurizer level has been restored and RCS pressure is equal to (or close to with the alternate criteria) the affected steam generator pressure. During natural circulation conditions, the upper head region could void. Since this would cause pressurizer level to increase rapidly, RCS depressurization is stopped on high level in the pressurizer to avoid a water solid condition. RCS depressurization is also stopped if pressure approaches saturation to avoid steam formation in the active RCS regions.

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